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HNPF RETIREMENT
DISPOSITION OF SODIUM

By

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I. INTRODUCTION

Two principal requirements for the retirement^{(1)*} of the Hallam Nuclear Power Facility (HNPF) were:

- 1) "Removal from the site of all sodium that can be drained from the system."
- 2) "Chemical reaction of residual sodium then remaining in the system, including all process lines, tanks, and the reactor vessel."

This report covers the plans, performance, and results of the operations accomplished in fulfilling these requirements. Since the overall retirement program covered a large scope of work, considerable planning was necessary to ensure that all tasks would be accomplished such that no health or safety hazard would be encountered either during or after the operations.

As a part of this planning, Atomics International (AI) prepared activity specifications covering the tasks to be accomplished and the requirements for each phase of the program. Consumers Public Power District (CPPD) prepared procedures detailing the manner in which each operation was to be performed in carrying out the various tasks. A list of all the activity specifications and operating procedures may be found in Appendix A. This report covers the removal of all bulk sodium from the HNPF site and the chemical reaction of residual sodium at the site.

*Superscript numbers in the text indicate references.

II. DRAINING THE REACTOR VESSEL

The reactor vessel is a cylindrical, stainless-steel, vessel approximately 18 ft in diameter and 40 ft deep with connecting piping as shown in Figure 1. The top surface or loading face is flush with the main floor of the reactor building. The vessel had been previously drained, as far as possible, by pumping the sodium out through the 3-in. fill-and-drain line and by using existing plant equipment, two linear-induction, electromagnetic pumps.

Due to the drain pipe configuration, the pumping operation left a sodium "heel" about two inches deep in the bottom of the vessel. In order to reduce the amount of residual sodium to be chemically reacted, it was desirable to remove as much of this "heel" as possible from the vessel. The 2-in. layer represented about 2400 lb of sodium. The problem was one of lifting the sodium to the top of the vessel, elevation 1440 ft 6 in., and then draining it to a part of the system from which it could be transferred to the storage tanks. In order to accomplish this, a pipe, extending to the reactor bottom, was installed through a fuel plug hole in the loading face shield. This pipe was then used to transport the sodium to the main floor level and to drain it into the sodium service system. The installation was designed to maintain the inert atmosphere in the reactor.

After a brief investigation of a specially designed lift pump similar to those used in wells, it was decided to use a combination of pressure inside the reactor and vacuum on the sodium service system to transfer the fluid. The design strength of the reactor bellows prohibited the use of pressure alone, and the depth of the reactor vessel prevented the use of vacuum alone. The routing of the drain flow was from the reactor vessel through a temporary drain pipe to the containment drain tank, Figure 2. From the containment drain tank, sodium was transferred to the primary sodium drain tank and then to the No. 1 primary sodium storage tank. Since the containment drain tank could not hold all the sodium, the operation was performed in steps.

First, a temporary drain line was installed from the reactor vessel, through a specially modified fuel plug, to the containment drain tank, see Figure 3. After the lines were preheated, a vacuum was drawn on the containment drain tank, and the reactor vessel was pressurized to establish flow; a

Figure 1. Sketch of Reactor Vessel and Connecting Piping

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pressure of 8.5 psig was required. When the containment drain tank was full, the reactor drain line was shut off and the containment drain was pressurized to transfer sodium through the primary sodium drain tank, to the primary sodium storage tank. This operation was repeated until the system ceased to function due to the low sodium level in the reactor vessel. The target level for the operation was 3/8-in. maximum sodium level. Final measurements indicated that the vessel was drained to a level of about 3/16 in., near the outside of the vessel bottom. The bottom of the reactor vessel was not perfectly flat; the center portion had a slight bulge making it higher than the outer portion.

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Figure 2. Routing of Flow for Draining Sodium "Heel" from Bottom of Reactor

Figure 3. Temporary Reactor Drain Line

III. REMOVAL AND SHIPMENT OF BULK SODIUM

A. GENERAL

The sodium heat-transfer systems and the sodium service systems were drained into the sodium storage (fill) tanks. These tanks were divided into two groups, primary sodium storage and secondary sodium storage.

The five primary sodium storage tanks were located in a below grade vault, floor elevation 1415 ft 0 in. at the southeast corner of the containment structure, see Figure 4. The primary sodium storage tanks are horizontal cylindrical tanks with dished heads at each end. The overall dimensions are 12 ft OD by 21 ft 3 in. end-to-end. The average sodium loading for the primary tanks was about 110,000 lb (1932 ft^3) each. Access to the primary tank vault was through a rectangular hatch 22 ft 6 in. long by 13 ft 11-1/2 in. wide in the main floor, elevation 1440 ft 6 in.

The three secondary sodium storage tanks were located in a separate underground vault, floor elevation 1420 ft 0 in., north of the primary vault. The secondary sodium storage tanks were also horizontal cylindrical tanks with dished heads at each end. The overall dimensions are 12 ft OD by 14 ft 5 in. end-to-end. The average sodium loading for the secondary tanks was about 76,660 lb ($1,330 \text{ ft}^3$) each. Access to the secondary tank vault was through a rectangular hatch, 17 ft long by 14 ft wide, in the main floor.

All of the tanks, primary and secondary, were equipped with electric heaters, thermal insulation, and metal sheathing. The heaters, insulation, and sheathing added about 1 ft to the diameter of each tank. The addition of heater terminal boxes added about 2 ft to the overall length of each tank. The tanks were set on supports made of structural steel. Since the terminal boxes made the primary tanks longer than the hatchway, a section of concrete was chipped out of one end of the primary tank hatchway.

Plans were made to disconnect the sodium storage tanks from the sodium service systems, remove them from the underground vaults, and load them onto railroad flat-cars for shipment to their destination. Electric heaters, thermal insulation, and heater terminal boxes were left on the tank. A detailed description of the removal plans is given in Reference 2.

Figure 4. Sketch Showing Location of Primary and Secondary Sodium Storage Tanks

The secondary tanks were shipped to Chatsworth, California, where they were transferred to trucks and brought to the Liquid Metal Engineering Center (LMEC) at the AI Santa Susana Facility. The primary tanks were shipped to PNL at Richland, Washington. The specific activity of the primary sodium was approximately $0.025 \mu\text{c/gm}$. The source of this activity was sodium-22 which has a 2.58-yr half-life.

Shipment of the sodium filled tanks was made on ICC Special Permit 4913 which permitted shipment in special liquid sodium storage tanks which did not meet the requirement of Section 73.206 of the ICC Regulations.

B. REMOVAL OF TANKS FROM UNDERGROUND VAULTS

Preparation for removal of the tanks started with turning off the electric heaters and allowing the sodium to cool and solidify, freezing point 208°F . After the tanks had cooled, all electrical, instrumentation, and piping connections were disconnected and removed. Piping nozzles on the tanks were sealed by welding. The main floor around the secondary tank vaults was shored up to ensure that there would be no danger of failure due to the heavy load of the tanks and handling equipment. A previous investigation had indicated that the reinforced concrete floor might not be adequate for these heavy loads.

Since the tank supports were not designed for dynamic loading, the supports were reinforced by the addition of structural steel members. The supports were then welded to the tanks, see p. 28 of Reference 2. The tank, support, and tie down arrangement for the railroad cars were designed for a shock load of 1 g in the fore and aft direction and for repeated loads of 0.5 g fore and aft, 0.2 g lateral, and 0.1 g vertical. The tank supports were also fitted with special lifting eyes to permit handling with cranes, see Figure 4 of Reference 2.

Discussions with the railroad shipping company had indicated that the dynamic loads could be maintained within the above limits by use of cushion underframe cars, by limiting train speeds to 40 mph on straight tracks and 30 mph on curves, and by limiting coupling and switching speeds to 1 mph.

The original removal plans had called for removal of the secondary tanks through the east wall of the building, or through the roof, once they had been lifted through the hatch; however, with his available equipment, the rigging contractor was able to slide these tanks south, over the primary tank vault and

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Figure 5. Secondary Sodium Storage Tank
Removed from Underground Vault
at HNPf

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Figure 6. Primary Sodium Storage Tank Being
Removed from Underground Vault
at HNPf

out of a roll-up door at the south end of the building where a railroad siding is located. In order to accomplish this, it was necessary to first remove some above grade equipment no longer required from the plant nitrogen and helium systems. Figure 5 shows the secondary sodium storage tank just after being removed from the underground vault.

The primary tanks were removed from the underground vaults in a manner similar to that for the secondary tanks. These were the heaviest tanks weighing about 74 tons each. Figure 6 shows the primary sodium storage tank being removed from the underground vault.

During the removal operation, some minor damage was sustained by the thermal insulation and sheathing. The damage was not too extensive to permit easy repair and operation of the electric heaters when the tanks reached their destination.

C. LOADING OF TANKS ON RAILROAD CARS

The general arrangement for loading of the sodium storage tanks on railroad flat-cars is shown in Figure 5 of Reference 2. The cars were previously prepared with oak cribbing and ballast for the primary tanks (not shown in Figure 5 of Reference 2) and the tanks were loaded as they were removed from the building; one tank was loaded onto each car. As previously mentioned, cushion underframe cars were used to reduce the dynamic loads during transit. Since the tanks were wider than the car beds, it was necessary to select cars with beds at least 47 in. above the rails to avoid contact with trackside docks and other obstructions enroute.

The cars used were 90-ton bulkhead flat cars, series 92500 to 92599, with a rail load limit of 263,000 lb. The cars were equipped with cushion underframes and antifriction bearings. For the primary tanks, the high center of gravity of the sodium-tank-plus-car-bed required the use of ballast, gravel loaded into cribs at each end of the car, to lower the center of gravity. About 34,300 lb of ballast was used on each car. Ballast was required to reduce the load eccentricity at the bearing between the wheel trucks and car bed when the car was sitting on a super-elevated curve. The ballast was added to keep this eccentricity less than one-third of the total bearing width.

The secondary tanks did not require lowering of the center of gravity; therefore, no ballast was used. Loading and shipping of the primary and secondary sodium tanks is shown in Figures 7 through 11.

D. SHIPPING OF TANKS

The cars with the sodium tanks were spotted at the forward end of the train, near the engine. This location was selected to minimize shock loads due to accumulative coupling slack during acceleration and deceleration. The train with the five primary sodium tanks, which went to Hanford, Washington, was accompanied by two messengers whose duty it was to inspect the cargo during stops and to give directions concerning the safe treatment and handling of the sodium tanks in the event of an accident. This last function would be particularly important in the event of a fire or rupture of a sodium tank.

No messengers were used on the train with the three secondary sodium tanks which went to Chatsworth, California. Inspection of the cars after arrival showed some slight loosening of the turnbuckles but not sufficient to permit load shift. The slight loosening was attributed to "seating" of the turnbuckle anchors to the car bed by vibration during transit. Since this was the first shipment made, additional precautions were taken with the primary tanks to ensure that all tie-downs were seated prior to shipment.

Figure 7. Secondary Sodium Storage Tank Being Anchored to Railroad Flatcar in Preparation for Shipment

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7709-1048

a.

7709-1012

b.

7709-1008

c.

7709-1010

d.

Figure 8. Typical Tiedown Arrangement and Details Used for Shipping Sodium Tanks

a.

7709-1060

b.

7709-1041

Figure 9. Loading of Primary Sodium Tanks Onto Flitcar, and the Ballast Arrangement

a.

7709-1061

b.

7709-1065

Figure 10. Rail Transport of Primary Sodium Tanks,
and Trackside Clearance

a.

7694-4022

b

7694-40214

Figure 11. Unloading Secondary Sodium Tank at Chatsworth, and Truck Transport to Santa Susana

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IV. DISPOSAL OF SECONDARY MAIN SODIUM PIPING

A. GENERAL

The disposal of the secondary main sodium piping was performed in accordance with HNPF Retirement Activity Specification No. 6, "Reaction of Residual Secondary Sodium and Retirement of Secondary Sodium Systems," and Retirement Detailed Procedure 6-1, "Isolation of the Secondary Sodium Service System and Disassembly of the System."

B. REMOVAL OF SECONDARY SODIUM PIPING

In preparation of separating the sodium piping from the secondary system, it was necessary to remove all insulation, heaters, and reflectors from the sodium pipes and secondary sodium system components. The insulation was buried on site after removal from the system. The reflectors and heaters were placed in the scrap yard.

The sodium pipe of the secondary sodium system was separated from the system components, such as the pumps, intermediate heat exchanger, valves, etc. before being cut into various lengths for removal from the pipe ways and system component module bay areas. As a section of pipe was separated from a system component, a plate was welded over the opening in the system component and a plastic cap was taped over the open end of the pipe to keep any sodium remaining in the system from reacting with the atmospheric moisture.

The pipe, after separation from a system component, was cut into various lengths. The length of each pipe section removed depended on the opening of the area where the pipe was being removed and the location in respect to the next component in this system; see Figure 12 which shows a typical loop and the approximate locations from which sodium pipe was cut.

The sections of pipe removed from the secondary sodium system were removed from the pipeways with the reactor high bay crane and stored in the reactor high bay until sufficient piping was available for cleaning.

Figure 12. Typical Sodium Loop Isolation Diagram

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C. CLEANING OF SECONDARY SODIUM PIPING

The secondary sodium piping that had previously been removed from the secondary sodium system and stored in the reactor high bay was transferred to an area south of the reactor building preparatory to the operation of cleaning each section of pipe.

The steam cleaning and sodium burning facility construction was incomplete at the time the secondary sodium system piping was available for cleaning; therefore, the secondary system piping was washed in a controlled area south of the reactor building on a concrete pad by Consumers Public Power District (CPPD) personnel. The personnel performing the operation wore flame retardant clothing and face shields.

A truck with an "A" frame hoist transferred one section of pipe from the storage area to the concrete pad where the plastic cap on one end of the pipe was removed. A fire hose was then used to flood the pipe from the open end. After the pipe was completely clean and the sodium had been reacted, the section of pipe was surveyed for contamination and then transferred from the concrete pad to the scrap yard as shown in Figure 13 to await final disposal.

All of the main secondary sodium piping was free of contamination after washing and sodium removal. The pipe was then available for sale as scrap.

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Figure 13. Main Sodium Piping in Scrap Pile After Cleaning

V. REACTION OF RESIDUAL SODIUM IN PRIMARY HEAT TRANSFER SYSTEM

The residual sodium in the primary, heat-transfer system was reacted with steam, in a nitrogen atmosphere. The operation was carried out in two parts. First, the residual sodium in the reactor vessel and piping from the vessel to the gallery seal was reacted. Next, the residual sodium in the primary heat transfer piping including the pump casings was reacted.

A. SELECTION OF REACTION

Since sodium is a relatively active metal, a number of chemical reactions were possible. The criteria for selection of a particular reaction process was that it had to be controllable, both in increasing and decreasing the reaction rate, and that the reaction products could be safely handled without damage to the reactor vessel or injury to personnel.

Control was a particularly sensitive problem since the reactor vessel contained some 7000 cubic feet of free space. Once this space is filled with a reactive agent or reaction product, controllability could be questionable.

Sodium reactions with oxygen (air), water (liquid and steam), carbon dioxide, and alcohols were investigated. Also, reactions with humidified air and humidified nitrogen were investigated, see Section IV of Reference 3.

The first experiments indicated that sodium reacts slowly with natural (no humidity controlled) air. In fact, the carbon dioxide in air might form a crust of sodium carbonate which would isolate the sodium from the air and stop the reaction before it was completed. Continued experimentation, see Appendix II of Reference 3, in this direction soon demonstrated the potential hazards concomitant with this method. In one of the tests, a 2-in. layer of sodium at the bottom of an 18-in. diameter cylindrical vessel was used. Atmospheric air was circulated through the vessel. After 19 days of a relatively slow reaction, with sodium hydroxide building up in a layer over the sodium, the vessel exploded; see Reference 7. The violence of the explosion indicated that it was apparently due to hydrogen detonation. Repeated tests using smaller sodium samples indicated that an explosion was highly probable if the depth of the original sodium layer is more than approximately 1/4 in.

Continued investigation disclosed the mechanism through which the explosive condition is created. The initial reaction at the sodium-air interface is a straight forward oxidation reaction with the oxygen of the air and the water vapor in the air.

As the reaction continues, the solid reaction products, Na_2O_2 and NaOH , build up in a layer over the sodium. Both of these compounds are hygroscopic and tend to absorb moisture from the atmosphere and form an aqueous solution. Once the aqueous solution covers the layer of sodium, the sodium is prevented from reacting with the atmosphere. At the air-hydroxide interface, more water is absorbed from the atmosphere, creating a dilute solution.

Air	Aqueous Hydroxide	Diluted Hydroxide	
Moist.	Layer	Solution	

The direction of the reaction depends on the temperature, concentration of the solution, and the vapor pressure of moisture in the atmosphere.

At the interface between the sodium and the sodium hydroxide solution, water from the solution reacts with the sodium and produces hydrogen which bubbles out through the solution to the atmosphere.

Diluted Hydroxide	Sodium	Concentrated Hydroxide
Solution	Layer	Solution

As indicated in Equations 5.3 and 5.4, the air side of the aqueous hydroxide layer is more dilute than the sodium side. This causes a concentration gradient and continual transfer of water from the air to the sodium. The reaction is slowed by a reduction of moisture in the atmosphere and may eventually stop if

the vapor pressure of the water over the hydroxide solution is greater than the vapor pressure of water in the air. In such case, water would leave both sides of the aqueous layer causing the hydroxide to cake into a crystalline mass; for example, if the hydroxide layer is at 40°C (104°F) with a concentration of 65 gm NaOH/100 gm H₂O at the air surface, the equilibrium water vapor pressure above this mixture is 13.2 mm Hg; see Table 3-27 of Reference 4. If the partial pressure of water vapor in the overlying air is less than 13.2 mm Hg, 25% relative humidity at 40°C, water evaporates from the hydroxide solution to the atmosphere; at the same time, water from the solution reacts with the underlying sodium and causes a net drying effect in the hydroxide layer.

The foregoing system is subject to a physical instability, especially if the original sodium layer is deep. The hydroxide solution is more dense than the underlying sodium; and during the reaction process, fragments or globules of sodium detach from the main body and float to the surface. This results in an increased reaction rate due to the increased sodium surface area and the higher water concentration at the surface of the hydroxide layer. The increased reaction rate is accompanied by an increase heat in production which is eventually sufficient to ignite the hydrogen when it escapes into the air.

The results of the previously mentioned tests using smaller amounts of sodium indicate that the maximum stable depth for sodium is about 1/4 in. This value may be subject to variation with temperature and humidity. Physical instability did not occur with those samples which were prepared in a manner permitting the hydroxide layer to drain away as it was formed; therefore, the reaction followed Equations 5.1 and 5.2.

Based on the test data and analysis of results, it was decided that the residual sodium in the HNPF primary system could be chemically reacted by using an inert gas and controlled humidity. In this manner, the risk of an explosion can be avoided. The most hazardous situation would be a rapid release of hydrogen and heat due to the physical instability problem.

Nitrogen was selected because of its relatively low cost. The nitrogen was heated and mixed with steam under temperature and pressure conditions which avoids condensation and subsequent addition of liquid into the reactor. The steam was readily available from the adjacent steam power plant. Further

protection against fire or explosion was provided by keeping the hydrogen concentration in the effluent nitrogen below that which would support combustion if the hydrogen-nitrogen mixture were mixed with air. The reaction was controlled by controlling the steam and nitrogen flow rates. Temperatures and pressures selected were within the safe operating capability of the equipment.

The efficacy of the steam-nitrogen process had previously been verified by tests in which a scaled down simulation of the reactor vessel was used, see Reference 5. These tests show that sodium layers with thickness similar to those which existed in the reactor vessel could be completely reacted by using steam-nitrogen mixtures with steam concentrations between 10 and 80% by volume. Also determined by the tests were that thick sodium layers, approximately 2 in. thick, are subject to physical instability and that this results in an increased reaction rate.

Other methods of chemically reacting sodium had been investigated and showed either practical or economic disadvantages when compared to the steam-nitrogen method. These disadvantages became apparent early in the studies obviating test work. Some of these methods and problems follow.

1) Reaction by Water Spray in a Nitrogen Atmosphere

Sodium deposits due to condensed sodium vapors were expected in areas of the reactor vessel such as on top of the loading face shield and around the Cerrobend seal; these would not be readily accessible to direct spray.

2) Reaction by Alcohol

Flooding of the reactor with alcohol would be expensive and would be a fire hazard due to the amount of combustible vapors. The fire hazard would be increased by the presence of hydrogen generated by the reaction. The use of alcohol vapors would require a complex recirculation system in order to conserve alcohol and to prevent the escape of fumes which are toxic and flammable.

3) Reaction by Dry Carbon Dioxide

High temperatures which are required for reaction with dry carbon dioxide would require the installation of extensive heating equipment. Such temperatures might also damage the reactor vessel.

4) Reaction by Moist Carbon Dioxide

This reaction produces sodium hydroxide as an intermediate product and sodium carbonate as a final product. The carbonate is not readily soluble in the hydroxide (Reference 6) and could form a crust which would seal the gas atmosphere from the remaining sodium.

B. PREPARATION OF HEAT TRANSFER SYSTEM

After draining most of the residual sodium from the reactor vessel, the reactor vessel and the three heat transfer loops were prepared for the steaming operation. First the primary sodium block, throttle, and check valves were removed. These valves were in the main loop piping between the reactor and the rest of the primary heat transfer system, see Figure 14. The valves were located on the east and west sides of the reactor, just outside of the diaphragm seals between the reactor cavity and the primary pipe gallery. A 9-ft x 14-ft hatchway in the main floor on each side of the reactor provided access for valve removal.

Modified pipe caps fitted with 2-in. stub ends for future connection to the steaming system were welded to the primary sodium piping where the valves had been removed, see Figures 15 and 16. Sodium pipes leading to and from the intermediate heat exchangers (IHX's) were cut and capped near the IHX locations. A 4-in. bypass line was installed between the cut end of each pump discharge line and the corresponding reactor return line, see Figures 17 through 20. Connections from the primary sodium pumps to the cover gas system and the balancing legs were also cut and capped. The pump covers were reinforced by welding structural steel to the cover plates and nozzles for attachment to the steaming system, and pressure gages were installed in the cover plates.

Six sodium sample holders, each having two cups for holding sodium samples, were fabricated and hung from selected fuel plugs in the loading face shield. These holders are shown in Figure 1 of Appendix C. Figure 2 of Appendix C shows the installation locations. Sodium samples were put in these holders for future inspection after the steaming operation.

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Figure 15. Main Sodium Piping in East Gallery
with Pipe Caps and Stub Ends Welded
Into Place

Figure 16. Main Sodium Piping in West Gallery
with Pipe Caps and Stub Ends Welded
Into Place

Figure 17 IHX Outlet and Pump Discharge Lines
Cut and Capped in Preparation for
Installation of Bypass Loop

Figure 18 Bypass Loop Piping Welded Into Place

Figure 19. Bypass Loop Piping with Heaters and Thermocouple Installed

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Figure 20. Completed Bypass Loop Installation

C. PREPARATION OF STEAMING SYSTEM

The steaming system was designed to supply a controlled steam-nitrogen mixture to the primary heat transfer system and to provide safe venting of the effluent gases to the plant exhaust stack. The steaming system was first connected to the reactor vessel; it was then modified and connected to the three primary heat transfer loops. For the reactor, the steam-nitrogen mixture was admitted to the connecting sodium pipes and the effluent gases were vented through the two reactor vent lines, see Figure 21. For the primary heat transfer loops, the steam-nitrogen mixture was admitted to the tops of the primary pump casings and vented from the pipe ends which were formerly connected to the reactor vessel, see Figure 22.

1. Piping and Equipment

In order to avoid excessive costs, existing plant equipment was used wherever possible. Components were selected on the basis of being "adequate" while not being specifically designed for the service for which they were used. The primary sodium service drain tank was cleaned and used as a mixing tank. One of the two spent fuel heat exchangers was used to condense steam from the effluent gases. Electrical heaters were salvaged from the sodium service systems for use on the steaming system pipelines. These heaters were connected to existing plant power supplies and control systems. Existing valves and pipe were also used where possible. A similar approach was used for the instrumentation, see Section C-2.

Purchased components were selected and obtained from standard industrial suppliers. The basis of selection was adequacy for the short time the components were to be in service. Another consideration in the selection of purchased components was delivery. Items requiring long delivery times were avoided. Substitutes or field expedients were used instead.

Functionally, the steaming system can be considered as being in two parts, the inlet subsystem and the effluent subsystem. The supply of steam and nitrogen was taken from existing pipelines located on the main floor, elevation 1440 ft 6 in., near the reactor vessel and was piped through the inlet system to the reactor sodium lines. The effluent gases were led from the reactor vent lines, through the effluent system, to the existing exhaust stack.

An overall view of the installed steaming system is shown in Figure 23. This photograph was taken while looking south and downward at the main floor in the reactor building. The reactor loading face shield is in the center foreground. The above grade portion of the inlet subsystem is on the left; the mixing tank and throttle station are near the column just south of the reactor. From the mixing tank, piping leads to access hatch 9A on the left side of the reactor and then to access hatch 9B on the right side of the reactor. The above grade portion of the effluent subsystem is on the right; the effluent condenser stands in front of the office windows. From the effluent condenser, lines lead through the south wall to the exhaust stack.

a. Reactor Inlet Subsystem

The steam supply was drawn from the existing 150-psig saturated steam system, which was originally used for heating and cleaning operations. A 1-in. pipeline led from the supply to a manual throttling station. From the throttling station, a 2-in. pipeline led to the mixing tank. The 1-in. steam line was equipped with an air operated valve that would close automatically on a reactor high pressure signal. The steam line contained suitable blowdown connections to permit heating and draining.

The nitrogen supply was drawn from the existing 50-psig nitrogen system which was originally used for cooling and flooding the underground vaults. The system normally had a 250,000-ft³ storage tank, and evaporators capable of handling about 200 ft³/min. A 60,000 ft³ reserve tank and additional evaporators were added to ensure an adequate nitrogen supply for the steaming operation between scheduled truck deliveries of nitrogen. A 2-in. pipeline led from the supply, through electric heaters, through the manual throttling station, and then to the mixing tank. The heaters were used to prevent condensation when the nitrogen was mixed with steam in the tank.

At the manual throttling station, both the steam and the nitrogen lines had two globe valves in parallel. One valve for each line was full size, the other was of a reduced size, see Figure 24. These valves were located near the steam and the nitrogen flowmeters to facilitate setting of the operating conditions.

From the mixing tank, the steam-nitrogen mixture was piped through electrically heated 2-in. lines to two 6-in. manifolds which were located below the

Figure 23. Overall View of Reactor Building Main Floor with Steaming System Installed

Figure 24. Manual Throttling Station

main floor in the pipe gallery on each side of the reactor shield. These were the locations where the main sodium valves were originally installed. Access plugs in the main floor provided a convenient routing for the pipelines from the mixing tank to the inlet manifolds, see Figures 25 and 26.

A 2-in. nitrogen line bypassed the electric heaters, the throttling station, the mixing tank, and connected to the lines leading to the inlet manifolds. The bypass line was equipped with a pressure regulator to maintain a positive pressure, approximately 1/2 psig minimum, in the reactor vessel. The purpose of this line was to prevent a vacuum from forming in, and drawing air into, the reactor vessel in the event of an accident. The results of such conditions could produce an explosive gas mixture.

Each 6-in. manifold had four 2-in. lines leading to the reactor sodium pipes, see Figure 27. The inlet manifolds and connecting lines were electrically heated to prevent steam condensation. A blowdown drain was provided on each manifold. A check valve and a manual shut-off valve were installed in each of the lines leading from the manifolds to the reactor sodium pipes in order to permit admission of the steam-nitrogen mixture through one sodium line at a time. The reactor sodium piping consisted of three 14-in. lines which connected to the reactor vessel below the core, three 16-in. lines which connected to the vessel above the core, one 6-in. line which connected to the vessel in the core region, and one 3-in. line (fill and drain) which ran through the vessel wall above the core and extended down to about 2 in. from the vessel bottom.

b. Reactor Effluent Subsystem

The conceptual design for the effluent subsystem showed the effluent gases being carried to the stack through the existing R/A vent and liquid waste systems, see page 31 of Reference 3; however, subsequent analysis of these systems indicated that there would be too much pressure drop, especially to handle an emergency pressure surge. In order to provide adequate flow capacity without requiring high reactor pressure, the reactor vent connections were piped directly to a condenser to remove the steam from the flow stream, and then to the stack. A manual, butterfly-type valve was used on the condenser outlet to maintain a small positive pressure on the reactor.

Figure 25. Hatch 9A on East Side of Reactor

Figure 26. Hatch 9B on West Side of Reactor

Figure 27. Manifold and Connecting Pipes in West Pipe Gallery

Figure 28. Effluent Condenser and Gas Analyzer Equipment

Figure 29. Typical Connection to Primary Pump Case During Installation

Figure 30. Typical Connection to Primary Pump Case After Installation Was Completed

The two 3-in. vent lines leading to the reactor vessel near the top were connected to the effluent subsystem. These lines had been used formerly as cover gas lines and were connected to the helium supply system and to the sodium service system. The vent pipes were cut and connected to 3-in. effluent lines which led up through the access openings to a 4-in. header; the 4-in. header then ran to a condenser. All of the effluent lines to the condenser were electrically heated to prevent condensation from running back into the reactor vessel.

A shell-and-tube type heat exchanger which was originally a heat exchanger for the spent fuel storage pit was used as the effluent condenser. The heat exchanger was set up at an approximately 6° angle for drainage, and the effluent gases were run through the tube side, see Figure 28. The condenser was cooled by water from the plant cooling water system. A 2-in. cooling water line was run from a below-grade nitrogen cooler, up through a hatch near the condenser, and then returned to the cooling water drain by the same route.

Water condensed from the effluent gases was drained to the holdup and monitoring tanks for the laundry room and decontamination room. Noncondensable gases from the reactor effluent, principally nitrogen and hydrogen, were piped to the existing exhaust stack by a 4-in. line which ran the length of the main floor and up over the filter room roof. An additional drain to the monitoring and hold-up tanks was provided at the riser which ran up over the roof. The purpose of this drain was to prevent any additional condensation from collecting in the effluent piping and causing a back-pressure on the system.

c. Heat Transfer Loop Inlet Subsystem

For steaming the primary heat transfer loops, the line formerly leading from the mixing tank to the 6-in. inlet manifolds was disconnected and run to the tops of the three primary pump casings where it was connected to the cover plates, see Figures 29 and 30.

d. Heat Transfer Loop Outlet Subsystem

The inlet manifolds were converted to an outlet manifold by connecting them to the six sodium pipes which led to the pump suctions and the IHX outlets, see Figure 21. By this time, the IHX's had been disconnected from the sodium piping, and bypass lines were installed, see Section B. Since the flow in the lines

connecting the sodium pipes to the manifolds would now be reversed, the poppets were removed from the check valves in these lines. The lines which formerly led the steam nitrogen mixture to the manifolds were disconnected from the inlet subsystem and connected to the two 3-in. effluent lines which had been connected to the reactor vents.

A comparison of the changes made to convert the steaming system from reactor steaming service to the loop steaming service can be seen by comparing Figures 21 and 22.

In order to permit checkout of the system and to permit operators to practice system operation, a 2-in. bypass line was installed between the inlet subsystem and the effluent subsystem. The piping and valve arrangement permitted steam and nitrogen to be run through the bypass line instead of through the reactor vessel, or the heat transfer loops. During this time, the reactor vessel, or heat transfer loops, could be maintained under slightly positive nitrogen pressure by use of the nitrogen line which bypassed the heaters, see Section V-C-1-a.

2. Instrumentation

Most of the instrumentation used to monitor the process of chemically reacting the residual sodium in the reactor vessel and three heat transfer loops (excluding the IHX's) had been salvaged from the HNPF facility or was borrowed from others. This reduced the cost of the system and eliminated additional capital investment.

A description of the use and origin of each instrument is given in Table 1. Those instruments which were part of the facility, such as the reactor vessel preheat temperature indicators and controls, are not listed.

The list of instruments, in general, is similar to the process system flow as shown in Figure 21; that is, in the following sequence.

- 1) Nitrogen supply to the heaters and mixing tank
- 2) Steam supply to the mixing tank
- 3) Steam-nitrogen mixture from the mixing tank to the reactor vessel
- 4) Effluent gas from the reactor to the condenser
- 5) Noncondensable gas from the condenser to the stack

Table 1. Instrumentation List

Instrument No.	Parameter	Description	Source
FI-900	N ₂ supply flow rate	Brooks, 1-1/2-in., variable-area, flow-through flowmeter; 0 to 150 scfm N ₂ at 60°F and 50 psig	HNPF
TI-5	N ₂ temperature at heater outlet (H-2)	U.S. -Gauge, well-type thermometer; 50 to 350°F	HNPF
TIC-1	N ₂ heater temperature, at controller	Indicating temperature-controller built into heater assembly; controller and protective device for H-1	AI*
TI-6	N ₂ temperature at heater outlet (H-1)	U.S. -Gauge, well-type thermometer; 50 to 350°F	HNPF
PI-14	N ₂ pressure at H-1 outlet	Ashcroft, Bourdon-tube type pressure gauge; 0 to 100 psig	HNPF
FI-2	N ₂ flow at mixing tank inlet (T-1)	Minn. Honeywell, Model 292D-15, self-indicating, DP cell with SS bellows; 0 to 100 in. water input; square root scale 0 to 100% flow; formerly FI-916 at HNPF	HNPF
		2-in., 300-lb, orifice flanges with opposing flange taps, and 2-in., Sch. 40, connecting pipes; formerly FI-605E at HNPF	HNPF
		3/32-in., thick orifice plate with 1.114-in. sharp-edged concentric orifice	N.D.†
TI-1	Steam temperature at mixing tank inlet	Well-type thermometer; 50 to 550°F, Ashcroft, 4-in. dial, local readout, 2-in. deep well, 3/4-in. NPT	HNPF
PI-13	Steam pressure at steam line to mixing tank	Ashcroft, Bourdon-tube type pressure gauge; 0 to 100 psig	HNPF
FI-1	Steam flow to mixing tank	Minn Honeywell, Model 292D-15, self-indicating DP cell with SS bellows; 0 to 100 in. water input; square root scale 0 to 100% flow; formerly FI-628 at HNPF	HNPF
		2-in., 300-lb, orifice flanges with opposing flange tops; and 2-in., Sch. 40, connecting pipes; formerly FI-631E at HNPF	
		3/32-in. thick orifice plate with 1.260-in. sharp-edged concentric orifice	N.D.†
PI-2	Pressure on mixing tank	Ashcroft; SS, Bourdon-tube 30-in. Hg; vac 0 to 30 psig, 1/4-in., MPT Mtg; installed with condensation coil and valve; both on hand at HNPF	N.A.§
PI-3	Pressure at reactor vessel inlet manifold	Ashcroft, SS, Bourdon-tube, 30-in. Hg, vac 0 to 15 psig, 1/4-in. MPT Mtg; installed with condensation coil and valve and Barksdale press; SW model D2H-H18 at HNPF	N.A.§
PI-5	N ₂ mixing tank bypass pressure	Ashcroft, Brass-bellows press-gauge; 0 to 5 psig	HNPF
TR-1	Steam/N ₂ Inlet, east manifold, temperature	Leeds-and-Northrup, Model-H, 6-in. strip-chart recorder; 0 to 400°F scale; IC-TC input, single-pen, 3-in./hr chart speed; formerly NLR-5; converted	HNPF
TR-2	Steam/N ₂ inlet west manifold temperature	Leeds-and-Northrup, Model-H "G" strip-chart recorder, 0 to 400°F; IC-TC input, single-pen, 3-in./hr chart speed; formerly NLR-3; converted	HNPF
TRC-2	Reactor vessel base temperature	Leeds-and-Northrup, Model-G, strip-chart recorder, 0 to 1200°F; IC-TC input, 16-pt alarm-to-annunciator, main control room, formerly IRC-P-10	HNPF
TRC-1	Reactor vessel effluent temperature	Leeds-and-Northrup, Model-H, strip-chart recorder, 0 to 800°F, IC-TC input; formerly NLR-4; converted	HNPF
PRC-1	Reactor vessel effluent line pressure	Mason-Neilan, Model-62100-18, 30-in. Hg vac; 0 to 15 psig; input; SS-Bourdon tube, self-indicating with 3 to 15 psi transmitter; formerly PI-118T at HNPF	HNPF
		3 to 15 psig, Model-5430F-E, Foxboro, 4-in. strip-chart recorder, 100%, 0 to 100% linear scale, 30-in. Hg, 0 to 15 psig; formerly TR204-205-206	HNPF
		Barksdale, D2H-H18; 2 required; from HNPF; Na throttle valve actuators	HNPF
FR-3	Reactor vessel effluent flow	Minn-Honeywell, Model-292N8-G6, diff press cell; 5 to 25 in. H ₂ O input, with 3 to 15 psig transmitter; formerly DPI-900T	HNPF
		3 to 15 psig receiver, Bourdon-tube type, 0 to 100%, square square-root marked, 3-in. scale, formerly installed on main board	HNPF
		4-in., 300-lb orifice flanges with opposing flange taps; and 4-in., Sch. 40, connecting pipes; formerly FI-615E at HNPF	New
		5/32-in. thick orifice plate with 1.348-in. sharp-edged concentric orifice	HNPF
TI-7	Reactor vessel effluent temperature	Ashcroft, well-type thermometer; 50 to 300°F	HNPF
PI-7	Condenser inlet cooling water press	Ashcroft, Bourdon-tube type press gauge; 0 to 100 psi	HNPF
TI-4	(Condenser outlet) effluent gas temperature	Well-type thermometer; 30 to 180°F; US gauge, 4-in dial with SS well; formerly TI-604	HNPF
FI-4	(Condenser outlet) effluent gas flow	Fisher-and-Porter, sight-flow ind with modified flapper; formerly FI-609 at HNPF	HNPF
PI-4	Effluent gas line to stack pressure	Ashcroft, SS-Bourdon-tube type, 30-in. Hg vac to 15 psig, 4-in. dial	New
TI-2	Condenser cooling water inlet temperature	Well-type thermometer; 30 to 180°F, US gauge, 4-in. dial with SS well; formerly TI-606	HNPF
TI-3	Condenser cooling water outlet temperature	Well-type thermometer, 30 to 180°F; US gauge, 4-in. dial with SS well; formerly TI-617 at HNPF	HNPF
LI-1	Condensate level (from condenser) in LC-1	Magnetrol, Model-27, float-operated level-Sw; sight-glass level indicator	HNPF
LI-2	Condensate level (from line to stack) in LC-2	LC-2 same as LC-1 LI-2 same as LI-1	HNPF
Alarm System		Panelarm, 10-window unit; formerly on Board 10 at HNPF	HNPF
FI-5	Condensate flow from condenser	Barton, Model-306 Flotrac; positive displacement, read out in 1/1000 of a barrel (42 gal)	New
FI-6	Condensate flow from line to stack	Same as FI-5	New
H ₂	Hydrogen in reactor vessel effluent	Borrowed from SRE at AI where used to monitor the wash cell gas, analyzer. Complete system with pump, filter, flow indicator, hydration column, and Beckman hydrogen detector with a 0 to 5% scale	AI†
O ₂ Anal	Oxygen in reactor vessel effluent, N ₂ supply and steam	Beckman, dual-range, O ₂ analyzer, ranges 0 to 1% and 0 to 5% O ₂ ; formerly OA-900 at HNPF, converted to two inputs	HNPF

*AEC property on loan to HNPF

†New; Daniel Indiana

§New; Ashcroft

6) Condensate from the condenser to the monitoring and hold-up tanks

7) Oxygen and hydrogen gas analyzers.

Instrumentation for monitoring the steaming of the heat transfer loops was essentially the same as that used for the reactor vessel except for a few pressure indicators, one on each pump case vs one for the reactor vessel, and except for temperature indicators, the manifold temperature indicators were not used.

All instruments functioned as anticipated with no failure or erratic operation, except for the hydrogen analyzer. The hydrogen analyzer periodically recorded downward, low hydrogen concentration, spikes which could not be explained by any changes in the process. These "spikes" were of short duration and were followed by a recovery to the previous reading.

The steaming system was essentially manually controlled. Indicating instruments were monitored by the operators and control adjustments were made when necessary or scheduled. For safety, instruments measuring critical parameters were tied to automatic actions and alarms. The alarms and automatic actions are listed in Table 2 and are shown in Figures 21 and 22.

All automatic devices were equipped with manual overrides. Automatic closure of the steam and nitrogen inlet valves (SV-1V and SV-2V) required manual resetting before the valves could be opened again. The effluent valve (SV-3V) automatically re-opens after the condition causing closure is corrected (low effluent pressure). The air operated and solenoid valves were arranged such that failure of electrical or air power would result in the inlet valves failing-closed and the effluent valve failing-open.

In order to maintain a liquid seal between the condenser drain and the hold-up tanks, the controls for the automatic drain valve (SV-4V) were set to prevent the water in the drain tank (LC-1) from being completely drained; thus, interchange of atmosphere between the radioactive waste system and the steaming system was avoided. In the event of an electrical or air power failure, the SV-4V would have failed-open.

TABLE 2
ALARMS AND AUTOMATIC ACTIONS

Condition	Sensor	Alarm Location	Automatic Action
1. N ₂ inlet temperature to mixing tank below 350° F	TIC/1	TAL/1 Annunciator on main floor	None
2. Effluent outlet temperature above 450° F	TRC/1	TAH/1 Annunciator on main floor	None
3. Reactor vessel temperature below 250° F	TI/21-26 through TI/21-30	TAL/2 Annunciator in reactor control room	None
4. Steam-N ₂ mixture pressure to reactor below 1/4 psig	PI/3	PAL/3 Annunciator on main floor	None
5. Steam-N ₂ mixture pressure to reactor above 10 psig	PI/3	PAH/3 Annunciator on main floor	None
6. Effluent outlet pressure below 1/4 psig	PRC/1	PAL/1 Annunciator on main floor	Valve SV-3V on effluent line to stack closes
7. Effluent outlet pressure above 8 psig	PRC/1	PAH/1 Annunciator on main floor	None
8. Effluent outlet pressure above 10 psig	PRC/1	Indicator light on relay K2	Valves SV-1V and SV-2V on steam and N ₂ lines to mixing tank close
9. Water collecting in effluent line to stack	LC/2	LAH/2 Annunciator on main floor	None

D. PREPARATION OF EXISTING FACILITIES

During removal of the primary and secondary sodium storage tanks, it was necessary to remove the helium system control board and the helium supply manifold system. Since helium was no longer required as a cover gas, the cover gas system was pressurized from the nitrogen supply system by connecting the two low pressure tanks in parallel. The low pressure nitrogen tank (T-901) and the low pressure helium tank (T-801) were adjacent to each other

and both operated at 50 psig. Nitrogen was normally supplied to the low pressure nitrogen tank from an outdoor liquid nitrogen tank capable of storing 250,000 scf through an evaporator and a pressure control system.

It was originally planned to connect an auxiliary 45,000 scf supply of bottled nitrogen to the low pressure tanks for emergency use, in the event that the normal supply was exhausted or the evaporators frozen. Further investigation revealed that bottled nitrogen in this quantity would be awkward to handle, would require many bottles, and would necessitate installation of a manifold. Instead, an auxiliary 60,000 scf liquid nitrogen tank was installed adjacent to the 250,000 scf tank and an auxiliary set of evaporators were installed to increase the total evaporator capacity. The system was piped so that either or both tanks could supply nitrogen through both sets of evaporators.

The existing nitrogen pressure control system between the evaporators and the low pressure tanks was constructed with 3/4-in. pipe and could not pass the required 200 scfm for the steaming operation. In order to increase the flow capacity between the evaporators and low pressure tanks, a 2-in. bypass line with control valves was installed in parallel with the existing system.

E. STEAMING OPERATION

Prior to admitting steam to the reactor vessel, the operational status of the steaming system was thoroughly checked by using flow through the reactor bypass line. During this time, the reactor vessel was kept under a slightly positive pressure by the pressure regulator in the nitrogen line; this bypassed the heaters and throttling station. In addition to checking out the manual controls, the set points for alarms and automatic actions were also checked.

After making the necessary adjustments and correcting minor malfunctions in the steaming system, the reactor vessel was pressurized to 10 psig with nitrogen to check the hold-down devices on the fuel plugs. When all equipment functioned properly, the reactor was designated as ready to be steamed.

1. Steaming Reactor Vessel

The reactor steaming operation closely followed the procedures outlined in the AI Activity Specification No. 5 (Appendix B) and in the Retirement Detailed Procedure 5-5 (Appendix C). Steam and nitrogen flow rates entering the reactor

vessel were set to the "approximate" values given in the detailed procedure. Precise settings were not attempted since the flow rates as originally specified in the activity specification were based on approximate, rather than precise requirements. Occasional on the spot calculations were made to ensure that the steam-nitrogen mixture ratio and the total flow rate were in the correct range. All readings were logged for future analysis.

During operation, it was necessary to make some minor changes in flow rates and valve set positions in order to keep the reaction within desirable limits; i. e., steam flow rates were adjusted to keep the effluent hydrogen concentration under 5% which was the upper limit of the hydrogen analyzer readout.

As expected, the reaction proceeded in a well-controlled manner. No temperature or pressure surges were noted. At no time did the indicated oxygen content of the inlet nitrogen or effluent gases exceed 1%. The hydrogen meter showed some signs of erratic operation, as described in Section V-C-2, but the erratic readings were easily identifiable on the meter strip chart (see Figure 31). Generally, the hydrogen concentration in the effluent gas proved to be a good indicator of reaction rate and was fairly responsive to changes in inlet steam concentration.

When similar large scale steaming operations are to be performed, it is suggested that the effluent hydrogen concentration be considered as one of the prime indicators for control purposes.

One unexpected event occurred during the reactor steaming operation. After about 65 hr of steaming, the Cerrobend seal (a low melting temperature seal between the loading face shield and the reactor vessel) melted in a region about 10 ft long. Cerrobend is a bismuth-lead-tin-cadmium alloy, produced by the Cerro de Pasco Co., which melts at about 158°F. It was used in a manometer type seal around the periphery of the loading face shield. During refueling operations, it could be melted, permitting the loading face shield to be turned while maintaining a seal on the reactor vessel. The seal region was normally cooled during reactor operation by the loading face cooling system.

Approximately 1-1/2 in. of Cerrobend material was blown out of the seal in a local spot. In order to refreeze the seal, service air hoses were set up to direct an air stream across the Cerrobend surface and through the loading face

Figure 31. Copy of Hydrogen Analyzer Recording Strip Section

7709-5405

cooling coils. Another diagonally opposite melt occurred within a few hours. Service air was also used to cool this region. With the use of service air for cooling, it was possible to keep all the seal in a frozen or plastic state and to continue steaming; steam flow had been secured when the initial melt occurred.

The two melt regions had occurred approximately above the two reactor vent lines. It was postulated that the flow of hot steam and nitrogen through the vent lines had transferred sufficient heat to melt the seal to the vessel walls, by convection, and then to the seal region, by conduction. The steam-nitrogen mixture temperature flowing through these lines was about 300 to 350°F.

Steaming of the reactor vessel was continued for a total of about 82 hr, including 6 hr at the end, with zero hydrogen concentration indicated; after steaming, two sample cups were pulled from the reactor vessel. Both samples were in a "reddish-brown" liquid state. A small solid particle about the size of a match head was found at the bottom of one sample cup. This particle was put into water (to see if it was sodium) but no reaction occurred. It was then concluded that all residual sodium in the reactor vessel had been reacted with the steam and that drying operations could begin.

The drying operation was accomplished by setting the reactor preheaters up to 450°F and purging the reactor vessel with hot (~400°F) nitrogen at a flow rate of 80 cfm. The drying operation continued for about 38 hr after which the remainder of the sodium samples were pulled. Inspection of the samples indicated that the sodium hydroxide had solidified. The reactor vessel was then slowly cooled and all systems were secured except for one nitrogen line which maintained a low (~1/2 psig) positive pressure in the reactor vessel.

2. Steaming Heat Transfer Loops

After completion of the reactor steaming operation, the steaming system was modified, as shown in Figure 22, for steaming the heat transfer loops. The procedures for this phase of the steaming operation were given in the Retirement Detailed Procedure 5-6, which was also written in conformance with Activity Specification No. 5, Appendix D.

The primary heat transfer loops were steamed one at a time with the steam-nitrogen mixture entering each loop through the cover plate at the top of the pump case and effluent gases leaving through the lines leading to the pump

section and from the IHX, now isolated and bypassed. The steaming operation was essentially the same as that used for the reactor vessel except for the flow rates, which were reduced.

Response of the heat transfer loops to control valve adjustments was much more rapid and erratic than was that of the reactor vessel. The difference was probably due to differences in size and configuration of the two systems. The reactor vessel was essentially a huge pot with the major mass of sodium being on the bottom where steam was first admitted. Effluent gases left the vent lines which were near the top. There was adequate space (and time) for diffusion of hydrogen with the effluent steam and nitrogen. Also, all connecting piping and internal components drained to the bottom of the reactor vessel. The system had sufficient inertia to mask any rapid changes in temperature, pressure, or effluent hydrogen concentrations which might have occurred due to local reaction-rate transients.

On the other hand, the heat transfer loops were essentially pipe runs, each with significantly less volume (a few hundred cubic feet vs 7000 ft³) and less system inertia. All components drained to a point near the vent connections where local perturbations in reaction rate could produce significant changes in the effluent hydrogen indicator. The smaller volume of the heat transfer loops would also make pressure increases due to increased temperature and hydrogen generation rates more evident than in the reactor vessel. During the loop steaming operation, the indicated hydrogen concentration in the effluent gas was subject to sharp increases, sometimes exceeding the instrument range, 5%. The hydrogen concentration could be rapidly reduced by reducing or shutting off the steam supply while maintaining the nitrogen flow rate. Observation of the loop pressure, by use of a pressure gage on the pump case cover, and by monitoring the flow leaving the condenser by means of a glass-sided flow indicator, showed that sharp increases in hydrogen concentration were accompanied by pressure increases and white smoke in the condenser exit line. Pressure increases of approximately 2 psi were observed, but at no time did the loop pressure exceed, or approach, 10 psig which was set as an operating limit. The smoke observed in the "see-through" flowmeter subsided when the hydrogen concentration dropped.

Despite the fluctuating conditions, steaming of the primary loop was continued until each loop had been steamed, including 6 hr at a zero hydrogen concentration. After steaming of the three loops was completed, the loops were dried with the preheaters and hot nitrogen, and allowed to cool. A 4-in. diameter inspection hole was then cut into the pipe wall near the ends of the loop piping; each end of the loop piping was a low point in the sodium piping system. Inspection of the piping interiors showed that the residual sodium hydroxide had solidified into a crystalline mass with no trace of moisture evident. Samples of the residue were taken where an adequate amount remained in the pipe. These samples were subsequently mixed with water and tested for residual sodium. No traces of free sodium were found.

F. ANALYSIS OF RESULTS FROM REACTOR STEAMING OPERATION

1. Calculated Results

Inspection of the chemical equations for the reaction between sodium and steam (or water) indicates that two methods might have been used for determining the amount of sodium reacted in the reactor vessel. The basic reaction equation is

This equation may be expanded to

where:

x = amount of nitrogen entering and leaving the process (nitrogen served only as an inert diluent for the steam),

y = amount of steam entering the process,

z = amount of sodium reacted and sodium hydroxide generated,

$(y - z)$ = amount of steam leaving the process (due to process inefficiencies),
and

$(z/2)$ = amount of hydrogen generated.

During the process, measurements were made of x , y , $(y - z)$, and the quantity $\alpha = (z/2)/(z + 2)$ which is the hydrogen concentration at the condenser outlet. From the process equation, it would appear possible to calculate the amount of sodium reacted by using the steam and nitrogen inlets data plus the outlet hydrogen concentration:

$$z = 2 \left[\frac{\alpha}{\alpha - 1} \right] x .$$

Or, one could use the difference in amounts of steam entering and leaving the process:

$$z = y - (y - z) .$$

After reviewing the process and flow diagrams, and observing the process in operation, it was decided that the second method would give the least meaningful results for the following reasons.

- 1) The calculation would involve a small difference between two large numbers, making it subject to large errors due to small measurement errors.
- 2) A continuous blowdown was maintained on the steam line, after the meter orifice. This results in a portion of the steam inlet flow never reaching the reactor vessel.
- 3) All of the steam not used in the reaction did not pass through the condensate drain flowmeter. Some of the steam remained condensed in the sodium hydroxide. Even after drying, each sodium hydroxide molecule holds one or more molecules of water of crystallization.

Calculation of the amount of sodium reacted was therefore based on the effluent hydrogen concentration and the amount of nitrogen entering the system. Since the reaction process proceeded in a fairly steady manner and instrument readings were recorded at frequent intervals (1/2 to 1 hr), a quasi-static solution was assumed. The calculated reaction rate taken at the beginning and end of each time interval was averaged and was assumed to be constant over that interval; the nitrogen inlet flow rate was assumed to be equal to the nitrogen

outlet flowrate for each point. The results from each interval were totaled to get the final results.

Data from 95 points during the period from the start of steaming to the beginning of the 6-hr "zero-hydrogen" steaming (indicating that all sodium had been reacted) was used for the calculations. The calculations involved the use of compressible flow equations for orifice meters. A Fortran computer program was written to facilitate the numerical operations and to reduce the error which might have accumulated from hand calculations. The orifice flow equations were taken from Reference 8. A constant orifice discharge coefficient of 0.61 was assumed.

In addition to calculating the amount of sodium reacted, the computer program also included calculation of other parameters considered of significant interest during operation. A total of ten parameters was calculated for each data point.

- 1) Elapsed time in hours and minutes from the actual start of steaming operations
- 2) The steam inlet flowrate in lb/hr based on the inlet steam orifice reading
- 3) The nitrogen inlet flowrate in scfm based on the inlet nitrogen orifice reading
- 4) Volume percent of steam in the inlet gas mixture based on steam and nitrogen inlet flowrates
- 5) Total steam inlet in pounds based on the summation of average steam flowrates over the individual time interval
- 6) Volume percent hydrogen in the condenser effluent gas, a part of the input data
- 7) Sodium reaction rate in lb/hr based on the inlet nitrogen flowrate and the hydrogen concentration in the condenser effluent gas
- 8) Process operating efficiency in percent based on the calculated molal rate of steam reacting with sodium and the indicated rate of steam entering the reactor vessel, defined as zero for zero steam flow

- 9) The amount of sodium reacted in pounds since the start of steaming operations based on the average sodium reaction rate calculated for each time interval
- 10) Total process efficiency in percent based on the calculated moles of sodium reacted and the calculated total steam inlet to the reactor since the start of steaming operations

Items 2), 4), 5), 8), and 10) are subject to some error due to the steam blowdown flow leaving the system after the inlet steam orifice meter. Derivation of the equations used in the program and a program listing are given in Appendices D-1 and D-3.

The computer program was planned to read the input data for one point, perform the necessary calculations, and then print the results; as opposed to reading all the data and then performing the calculations. Only those calculated values which were necessary for averaging or summing were retained for use with calculations on the next data point. The process operating efficiency was arbitrarily defined as zero for those few points where there was zero steam flow but a measurable hydrogen concentration was still indicated in the effluent gas. A check was also included to stop the program in the event of a calculated negative time interval which could occur if the data cards were loaded in the wrong sequence.

All data, shown in Appendix D-4, for calculation of the results was copied from data recorded by CPPD. A complete copy of the original data was not reproduced herein because most of the information recorded was used solely for monitoring the status of the reactor and steaming system during the steaming operation and during preoperation checks.

During the steaming operations, few readings were taken of the pressure on the upstream side of the steam and nitrogen inlet orifices. That these pressures were not monitored at every point was an oversight; the gages had been installed immediately before the start of operations and the identification numbers were left off of the data sheets inadvertently. After the steaming operation was completed, a short test was run to obtain additional data for correlating the mixing tank pressure reading with the flow meter readings and the orifice pressure readings. These test data, along with the readings which had been

taken during the reactor and loop steaming operations, were used to establish an empirical relationship between the three readings. This relationship was then used for the main calculations. Derivation of the relationship is shown in Appendix D-2.

The printout of results from the computer calculations are shown in Table 3. Calculations indicate that a total of 483 lb of sodium were chemically reacted with steam in the reactor vessel and connecting piping. An earlier run had indicated that approximately 700 lb of sodium had been reacted, see Reference 9; however, a subsequent review of the calculation procedure disclosed the fact that the square root scale on the indicators for the steam and nitrogen flowmeters had been interpreted as being a linear scale. The computer program was then corrected and run again to give the present results.

Comparison of the process efficiencies in Columns 8 and 10 of the results printout, Table 3, show that the overall process efficiencies indicated in the earlier tests with a simulated reactor vessel were similar in magnitude, see page 6 of Reference 5. In fact, the reactor results data are practically equal to those for the longest, 16-hr, test. Higher process efficiencies, in the neighborhood of 18%, were found for rather steady operating intervals near the beginning of the reaction at Hallam. The calculated efficiency of 47% for the second point is misleading due to the previously high steam flow which was still affecting the hydrogen concentration. The initial high process efficiencies were due to the close proximity of sodium to the inlet line. Steam was first admitted to Line 41 which leads down, inside the reactor vessel, to within 2 in. of the bottom, near the periphery. The deepest portion of the sodium pool, $\sim 7/32$ in., was just under the pipe outlet.

No calculations were attempted for the amount of sodium reacted in the primary heat transfer loops, due to the wide swings in effluent hydrogen concentration, some of which were beyond the instrument range of 5% and the frequency with which manual controls had to be adjusted.

2. Inspection Results

After each steaming operation, reactor steaming and loop steaming, samples of sodium were taken from the primary system and inspected. Sodium samples were taken from the primary heat transfer loop piping by cutting a

TABLE 3
HNPF RETIREMENT RESULTS FROM STEAMING OF REACTOR VESSEL
(Sheet 1 of 3)

HNPF RETIREMENT
RESULTS FROM STEAMING OF REACTOR VESSEL

1 ELAPSED TIME	2 STEAM FLOW RATE	3 NITROGEN FLOW RATE	4 VOLUME PERCENT SIM.	5 TOTAL STEAM INLET	6 VOLUME PERCENT HYD.	7 SODIUM REACTION RATE	8 PROCESS OPER. EFF.	9 SODIUM REACTED	10 TOTAL PROCESS EFF.
HR/MIN	LB/HR	SCFM		LB.		LB/HR	PERCENT	LB.	PERCENT
0/ 0	117.340	105.212	28.13	0.0	2.500	19.646	13.103	0.0	0.0
0/ 28	46.920	104.690	13.59	38.3	3.600	28.471	47.488	11.227	22.925
1/ 0	57.246	104.713	24.58	76.8	3.050	23.990	19.306	25.217	25.706
1/ 30	100.381	125.436	21.93	126.2	3.200	30.197	23.543	38.763	24.043
2/ 0	100.114	126.244	21.77	176.3	3.000	28.433	22.227	53.421	23.714
2/ 30	100.187	126.223	21.79	226.4	2.800	26.479	20.684	67.149	23.214
3/ 0	57.035	105.542	24.39	275.7	2.900	22.955	18.514	79.507	22.571
3/ 30	57.035	104.491	24.58	324.2	3.000	23.534	18.981	91.130	21.998
4/ 0	57.045	104.432	24.59	372.7	2.900	22.715	18.317	102.692	21.562
4/ 30	93.264	87.455	27.23	420.3	2.900	19.021	15.961	113.125	21.064
5/ 0	90.455	67.618	31.95	466.2	3.400	17.331	14.995	122.213	20.515
5/ 30	91.703	76.612	29.58	511.8	3.700	21.436	18.294	131.905	20.171
6/ 0	52.133	77.792	29.36	557.7	3.750	22.072	18.749	142.782	20.035
6/ 30	91.497	77.167	29.38	603.6	3.700	21.591	18.468	153.698	19.927
7/ 0	91.508	77.127	29.40	649.4	3.330	19.348	16.547	163.933	19.756
7/ 30	91.302	65.880	31.44	695.1	3.100	16.280	13.955	172.840	19.460
8/ 0	90.946	69.783	31.38	740.6	3.260	17.125	14.736	181.191	19.146
8/ 30	114.841	68.633	37.00	792.1	3.180	16.416	11.187	189.576	18.731
9/ 0	139.695	69.428	41.39	855.7	3.270	17.092	9.575	197.953	18.104
9/ 30	139.711	69.392	41.40	925.6	3.500	18.328	10.267	206.808	17.486
10/ 0	100.009	68.801	33.78	985.5	3.650	18.980	14.853	216.135	17.164
10/ 30	99.947	68.809	33.76	1035.5	3.580	18.605	14.568	225.532	17.045
11/ 0	99.971	68.651	33.82	1085.5	3.350	17.328	13.565	234.515	16.908
11/ 30	102.189	70.049	33.86	1136.0	3.250	17.136	13.123	243.131	16.749
12/ 0	139.938	70.065	41.21	1196.6	3.100	16.323	9.129	251.496	16.449
12/ 30	142.351	71.314	41.19	1267.1	3.000	16.062	8.830	259.592	16.033
13/ 0	154.640	72.088	42.98	1341.4	3.050	16.515	8.347	267.736	15.620
13/ 30	192.489	70.312	49.00	1428.3	3.100	16.381	6.660	275.960	15.121
14/ 0	151.807	70.356	48.89	1524.3	3.300	17.485	7.134	284.427	14.603
14/ 30	191.139	70.166	48.87	1620.1	3.450	18.258	7.476	293.362	14.171
15/ 0	190.870	70.166	48.84	1715.6	3.550	18.807	7.711	302.629	13.805
15/ 30	190.870	70.210	48.82	1811.0	3.550	18.819	7.716	312.035	13.484

TABLE 3
HNPF RETIREMENT RESULTS FROM STEAMING OF REACTOR VESSEL
(Sheet 2 of 3)

HNPF RETIREMENT RESULTS FROM STEAMING OF REACTOR VESSEL									
1	2	3	4	5	6	7	8	9	10
ELAPSED TIME	STEAM FLOW RATE	NITROGEN FLOW RATE	VOLUME PERCENT STM.	TOTAL STEAM INLET	VOLUME PERCENT HYD.	SODIUM REACTION RATE	PROCESS OPER. EFF.	SODIUM REACTED	TOTAL PROCESS EFF.
HR/MIN	LB/HR	SCFM		LB.		LB/HR	PERCENT	LB.	PERCENT
16/ 0	189.695	69.865	48.79	1906.1	3.600	19.000	7.839	321.490	13.199
16/ 30	189.699	69.909	48.78	2001.0	3.550	18.738	7.731	330.925	12.943
17/ 0	151.170	70.504	48.76	2096.2	3.350	17.796	7.285	340.058	12.696
17/ 30	189.320	69.858	48.75	2191.3	3.150	16.546	6.840	348.644	12.451
18/ 30	190.928	69.278	49.16	2381.5	2.700	14.000	5.738	363.917	11.959
19/ 30	191.757	69.529	49.18	2572.8	2.500	12.983	5.299	377.408	11.480
20/ 30	283.983	84.774	54.03	2810.7	2.200	13.887	3.827	390.843	10.883
21/ 33	258.013	102.459	46.91	3095.2	1.680	12.749	3.867	404.827	10.236
22/ 30	388.390	124.133	52.34	3402.3	1.200	10.979	2.212	416.098	9.571
23/ 30	467.550	141.661	53.67	3830.2	1.000	10.420	1.744	426.798	8.721
24/ 30	405.653	144.819	49.57	4266.8	0.650	6.900	1.331	435.458	7.987
25/ 30	432.747	144.762	51.20	4686.0	0.550	5.830	1.054	441.823	7.379
26/ 30	445.483	142.936	52.24	5125.1	0.400	4.180	0.734	446.829	6.823
27/ 30	443.038	142.817	52.12	5569.4	0.300	3.129	0.553	450.484	6.330
28/ 30	443.242	143.811	51.96	6012.6	0.200	2.099	0.371	453.098	5.898
29/ 30	446.167	148.065	51.40	6457.3	0.190	2.053	0.360	455.173	5.517
30/ 30	438.917	151.986	50.33	6899.8	0.090	0.997	0.178	456.698	5.180
31/ 30	460.709	152.536	51.45	7349.6	0.150	1.669	0.283	458.031	4.877
32/ 30	459.351	152.180	51.44	7809.6	0.050	0.554	0.094	459.143	4.601
33/ 30	458.989	154.193	51.09	8268.8	0.100	1.124	0.192	459.982	4.354
34/ 30	477.970	153.588	52.20	8737.3	0.050	0.560	0.092	460.824	4.128
35/ 30	450.542	154.578	50.56	9201.5	0.050	0.563	0.098	461.385	3.924
36/ 30	413.437	145.358	49.95	9633.5	0.120	1.272	0.241	462.302	3.756
37/ 30	404.468	83.493	62.96	10042.5	0.100	0.609	0.118	463.243	3.610
38/ 30	522.344	66.096	73.87	10510.9	0.400	1.933	0.284	464.513	3.459
39/ 30	704.927	63.553	79.56	11129.5	0.800	3.732	0.414	467.346	3.286
40/ 30	768.943	64.806	80.63	11866.5	0.820	3.902	0.397	471.163	3.107
41/ 30	0.0	69.035	0.0	12250.9	0.500	2.526	0.0	474.377	3.030
42/ 30	0.0	68.323	0.0	12250.9	0.250	1.247	0.0	476.264	3.042
43/ 30	0.0	67.875	0.0	12250.9	0.0	0.0	0.0	476.887	3.046
44/ 30	317.267	61.751	64.32	12409.6	0.0	0.0	0.0	476.887	3.007
45/ 30	308.654	61.990	63.60	12722.5	0.020	0.090	0.023	476.933	2.934

TABLE 3

HNPF RETIREMENT RESULTS FROM STEAMING OF REACTOR VESSEL
(Sheet 3 of 3)

HNPF RETIREMENT
RESULTS FROM STEAMING OF REACTOR VESSEL

1	2	3	4	5	6	7	8	9	10
ELAPSED	STEAM	NITROGEN	VOLUME	TOTAL	VOLUME	SODIUM	PROCESS	SODIUM	TOTAL
TIME	FLOW	FLOW	PERCENT	STEAM	PERCENT	REACTION	OPER.	REACTED	PROCE SS
	RATE	RATE	SIM.	INLET	HYD.	RATE	EFF.		EFF.
HR/M IN	LB/HR	SCFM		LB.		LB/HR	PERCENT	LB.	PERCENT
46/ 30	306.417	60.701	63.92	13030.1	0.100	0.442	0.113	477.199	2.866
47/ 30	457.218	60.755	72.53	13411.9	0.110	0.487	0.083	477.664	2.787
48/ 30	508.700	61.884	74.67	13924.8	0.110	0.543	0.075	478.179	2.687
49/ 30	541.047	50.041	79.14	14479.7	0.100	0.365	0.053	478.633	2.587
50/ 30	542.146	46.508	80.36	15071.3	0.160	0.543	0.078	479.086	2.496
51/ 30	548.999	46.657	80.50	15566.9	0.080	0.272	0.039	479.494	2.411
52/ 30	553.236	46.873	80.55	16118.0	0.080	0.273	0.039	479.766	2.330
53/ 30	554.869	47.276	80.46	16672.1	0.060	0.207	0.029	480.006	2.253
54/ 30	457.772	36.053	81.67	17178.4	0.050	0.131	0.022	480.175	2.188
55/ 30	459.813	35.842	81.82	17637.2	0.050	0.131	0.022	480.306	2.131
56/ 30	457.994	35.842	81.77	18096.1	0.050	0.131	0.022	480.437	2.078
57/ 30	456.131	35.761	81.74	18553.1	0.050	0.130	0.022	480.567	2.027
58/ 30	523.929	50.297	78.52	19043.2	0.050	0.183	0.027	480.724	1.976
59/ 30	483.678	48.415	77.81	19547.0	0.0	0.0	0.0	480.816	1.925
60/ 30	518.611	50.025	78.44	20048.1	0.0	0.0	0.0	480.816	1.877
61/ 30	522.296	48.448	79.09	20568.6	0.0	0.0	0.0	480.816	1.829
62/ 30	518.615	47.784	79.20	21089.0	0.050	0.174	0.026	480.903	1.785
63/ 30	511.463	46.120	79.56	21604.1	0.020	0.067	0.010	481.023	1.743
64/ 30	494.517	47.296	78.58	22107.0	0.0	0.0	0.0	481.057	1.703
65/ 30	498.402	46.848	78.87	22603.5	0.0	0.0	0.0	481.057	1.666
66/ 30	511.858	47.170	79.20	23108.6	0.0	0.0	0.0	481.057	1.629
67/ 30	541.839	46.395	80.39	23635.5	0.040	0.135	0.020	481.125	1.593
68/ 30	557.706	47.246	80.55	24185.3	0.040	0.138	0.019	481.261	1.557
69/ 30	557.118	47.214	80.55	24742.7	0.170	0.585	0.002	481.623	1.523
70/ 30	558.631	47.209	80.59	25300.5	0.150	0.516	0.072	482.174	1.491
71/ 30	558.215	45.324	81.21	25859.0	0.100	0.330	0.046	482.597	1.461
72/ 30	558.312	47.186	80.59	26417.3	0.050	0.172	0.024	482.848	1.430
73/ 30	558.021	46.252	80.89	26975.5	0.030	0.101	0.014	482.985	1.401
74/ 30	553.529	47.226	80.44	27531.2	0.010	0.034	0.005	483.052	1.373
75/ 30	557.144	46.345	80.84	28086.6	0.010	0.034	0.005	483.087	1.346
76/ 30	558.548	46.869	80.70	28644.4	0.0	0.0	0.0	483.103	1.320

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4-in.-diameter hole in the top of the pipe wall and scraping out a sample. The sample holes were subsequently welded closed. The sample location for the loop piping was at the ends which connected to the steaming system. This was a low point to which liquids would drain. Since scraping for a sodium sample in the reactor vessel was impractical, sodium samples were placed in special containers and suspended inside the reactor vessel prior to the steaming operation, see Section VI of Appendix B. The samples were removed and inspected after completion of various stages of the steaming operation.

a. Reactor Vessel Results

The first two sodium sample sets, 2 cups, were removed from the reactor vessel after steaming for 6 hr with zero hydrogen indication in the effluent gases. The following is a quotation from the operators log book on the condition of sample in each cup:*

"Both samples were identical in appearance when removed from the reactor. The samples were in a liquid state and a reddish-brown color. They began to set up immediately into a bluish gelatin appearing semi-solid. Before setting up, both samples were probed for particulates. Only one solid piece of solid matter was found, that being in the bottom portion from C-128. This particle about the size of a match head was put into a water sample and no reaction occurred."

Based on the lack of evidence of unreacted sodium in the sample and the 6-hr of steaming with zero hydrogen indicated in the effluent gases, it was concluded that all residual sodium in the reactor vessel had been reacted chemically. The vessel preheaters were then turned up to 450°F and a 24-hr purge with hot 350 to 400°F nitrogen started.

At the end of a 24-hr period, a sample of the effluent nitrogen was cooled from 468 to 74°F, and the humidity was measured. The relative humidity was less than 2% at 74°F. This corresponds to a relative humidity of less than 0.002% at the effluent temperature of 468°F. The four remaining sample sets

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*Consumers Public Power District, Sheldon Station; Reactor Steaming System Log, entry at 0500 on April 12, 1958

were then removed and inspected. All samples had solidified and there was no evidence of unreacted sodium when they were mixed with water.

b. Primary Heat Transfer Loop Results

Each of the three heat transfer loops were steamed with 75% steam entering the loop until the effluent hydrogen concentration indicated zero for at least 3 hr. After the 3-hr period, the loops were dried by purging each with hot nitrogen until the relative humidity of the effluent nitrogen was less than 1%. After the loops had been dried, samples of the residue were taken from each of the six main pipe ends, two for each loop, where available. Two of the lines, No. 232 and 233, did not have enough residue for a sample. All samples taken were solid and showed no sign of unreacted sodium when they were mixed with water.

3. Potential Effects of Residue

The interior of the reactor vessel is considered to be a possibly dangerous source of radiation for a period of 100 years. In order to assure that the integrity of the vessel will be maintained for at least that period, an investigation was made of the potential effects of the sodium hydroxide on the vessel material. A complete description of the investigation and its results are given in Reference 10.

The principal conclusions of this investigation were as follows.

- 1) After completing the planned sealing operations for the vessel, it will be subject to a passive environment and no significant corrosion due to the caustic will occur.
- 2) The introduction of water or air into the vessel, by seepage or other means, will not measurably change the corrosion due to the caustic.
- 3) The low stress levels in the reactor vessel and the insignificant amounts of chlorides and sulfates in the residue are not sufficient to promote stress corrosion.

Based on the foregoing conclusions, it is estimated that the integrity of the reactor vessel will be maintained more than the required 100 years.

VI. DISPOSAL OF SODIUM SERVICE PIPING

A. REMOVAL OF SECONDARY SODIUM SERVICE PIPING

The preparation for the removal of the secondary sodium system was initiated with the removal of the insulation, heaters, and reflectors from the sodium piping and sodium system components. This facilitated separation of the sodium piping from the system components.

The secondary sodium service piping was separated from a system component by use of a pipe cutter at the stub joint. A stainless steel seal plate with a stainless steel bar stock valve was welded over the open stub joint to the system component. A plastic cap was then taped over the open end of the pipe.

A nitrogen gas line was connected to the stainless steel bar stock valve in the seal plate and the secondary sodium system was purged with nitrogen both to prevent the residual sodium in the system from reacting with the atmospheric moisture and to retain an inert atmosphere within the system.

The secondary sodium service piping was separated from all system components and then each section of pipe, if too long to remove from the pipe ways, was cut at the tees and bends to facilitate removal from the below grade areas to the reactor high bay for storage. The pipe hangers were removed and the system components were blocked in place as sections of pipe were removed and transferred from the pipe ways and to the system component bay to the reactor high bay.

B. CLEANING OF SECONDARY SODIUM SERVICE PIPING

The secondary sodium service piping was transferred from its storage area in the reactor high bay to the sodium disposal facility for cleaning after all the piping had been removed from the system.

Approximately 627 ft of secondary service piping was cleaned in the sodium disposal facility, Figures 32 and 33. After cleaning, the pipe was transferred to the pipe scrap pile south of the reactor building to be disposed of as clean scrap.

Figure 32. Sodium Disposal Facility Buildings

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Figure 33 Interior of Sodium Cleaning Building

C. REMOVAL OF PRIMARY SODIUM SERVICE PIPING

The primary sodium service piping was removed from the primary service system in accordance with HNPF Retirement Activity Specification No. 5, Reaction of Residual Primary Sodium and Retirement of Primary Sodium System; and detailed procedure RDP-5-1, Isolation of Primary Sodium Service System and Disassembly of System.

The primary sodium service piping was prepared for removal in the same manner as that for the secondary sodium system by removing the pipe and system component insulation heaters and reflectors to facilitate the cutting of the pipes first at the juncture of the system components and by cutting at pipe tees and pipe bends to facilitate removal for cleaning. As a section of pipe was separated from a system component, the opening to the component was closed by welding a stainless steel seal plate over the opening. A stainless steel bar stock valve in the seal plate was used in purging the system with nitrogen to retain an inert atmosphere within the system.

D. CLEANING OF PRIMARY SODIUM SERVICE PIPING

The dismantled primary sodium service piping was removed from the below-reactor-grade areas and was transferred to the sodium disposal facility for cleaning and reacting of residual sodium. There was approximately 4088 ft of primary service pipe cleaned and approximately 3800 lb of sodium removed from the pipe and reacted in the sodium disposal facility. All sections of the primary sodium service piping were surveyed for contamination, by the HNPF health and safety personnel; all of the pipe was found to be clean and was placed in the scrap metal yard.

VII. REMOVAL AND DISPOSITION OF MAJOR COMPONENTS

A. MAIN SODIUM BLOCK AND THROTTLE VALVES

The removal of the main primary sodium block and throttle valves was performed in accordance with HNPF Retirement Activity Specification No. 5, Reaction of Residual Sodium and Retirement of Primary Sodium System; and detailed procedure RDP-5-2, Removal of Primary Sodium Block and Throttle Valves and Isolation of the Reactor from its Related Systems. The main secondary sodium block and throttle valves were removed in accordance with HNPF Retirement Activity Specification No. 6, Reaction of Residual Secondary Sodium and Retirement of Secondary Sodium System; and detailed procedure RDP-6-5, Removal of Secondary Heat Transfer Piping and Components.

The main primary sodium block and throttle valves were located in the east and west pipe galleries just out of the reactor cavity below access plugs No. 9A and 9B.

The pneumatic valve operators, pipe hangers, and valve structural supports were removed to provide access for the pipe cutter used in separating the valves from the main primary sodium pipe lines. Temporary blocking was used to support the valves and piping until the valves could be removed.

The valve and pipe insulation, heaters, and reflectors were also removed to provide access for the pipe cutter. After separating the valves from the system, plastic caps were placed over the inlet and outlet openings of the valves. The reactor side and the system side of the sodium pipe was sealed by welding a stainless steel pipe cap over the open line. Each pipe cap was fitted with a 2-in. capped stub end for future connection to the steaming system.

The main primary sodium throttle valves were removed as an assembly with the main primary sodium check valve. The throttle and check valve were raised from the below-reactor-grade pipe-way to the reactor high bay by use of the high bay 60-ton crane. The throttle and check valve were then separated by use of a pipe cutter on the reactor high bay floor. There were three main primary sodium block valves and three main primary sodium throttle and check valves. After the valves were removed from the primary system and transferred to the reactor high bay, the plastic caps were removed from the valve

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Figure 34. Primary Sodium Block Valve After
Cleaning at Santa Susana

AI-AEC-MEMO-12736

inlet and outlets and were replaced by welding a stainless steel seal plate over the openings. The valves were then crated for shipment off site.

The main secondary sodium throttle valves were removed in a similar manner to that used for the main primary sodium valves. There were no block or check valves in the main secondary sodium system.

The three main primary sodium, block, check, and throttle valves together with the three main secondary sodium throttle valves were shipped to the Liquid Metals Engineering Center at Santa Susana, California. Figures 34 and 35 show the primary sodium block and throttle valves after arrival at Santa Susana.

B. STEAM GENERATORS AND RELATED COMPONENTS

The three steam generators, each consisting of an evaporator, a superheater, and an eliminator, Figure 36, were isolated from the sodium system and removed from the steam generator rooms as outlined in HNPF Activity Specification No. 6, Reaction of Residual Secondary Sodium and Retirement of Secondary Sodium System; and detailed procedure RDP-6-4.

The isolation of the steam generator system began in March 1968 with the isolation of the steam generators, superheaters, and evaporators from the secondary sodium system and the interconnecting pipe lines between each component. Steel seal plates were welded over each opening where pipe connection was separated from the component. One seal plate on each component contained a bar stock valve to facilitate leak checking of the welds of the seal plates.

It was necessary to remove the heating and ventilating ducting, ladders, and elevated catwalks from the steam generator rooms to facilitate the removal of the steam generators and system components. Wood cribbing-and-blocking was used to support the components until removed for shipping off site.

During the isolation of the superheater and evaporator vessels, cradle extension assemblies were installed on the vessels to provide support for the vessel during shipment.

The steam generators, superheaters, and evaporators were shipped by rail to Idaho Nuclear Corp. NRTS, Central Facilities Area, Scoville, Idaho for storage. The shipments were completed in June 1968.

7694-5463

Figure 35. Primary Sodium Throttle Valve After Arrival at Santa Susana

7508-5043

Figure 36. HNPF Steam Generator Assembly

C. INTERMEDIATE HEAT EXCHANGERS

The intermediate heat exchangers, Figure 37, were isolated from the reactor sodium systems as outlined in Activity Specifications No. 5, Reaction of Residual Primary Sodium and Retirement of Primary Sodium Systems; and No. 6, Reaction of Residual Secondary Sodium and Retirement of Secondary Sodium System. Also followed, were detailed procedures RDP 5-3, Isolate Intermediate Heat Exchanger Modules from their related Primary System; and RDP 6-2, Isolate the Intermediate Heat Exchangers from their related Secondary Systems.

The isolation of the IHX modules included cutting of the primary and secondary main heat transfer piping outboard of the inlet and outlet nozzles of the modules, welding of closure plates onto the remaining line stubs, and cutting and capping the vent lines which had been used to vent the secondary sodium side of the modules to the expansion tanks.

Bar stock valves were attached to the primary loop inlet and outlet nozzle closure plates and the secondary loop vent nozzles closure plates. These bar stock valves were installed during the module isolation to provide a means of maintaining an inert gas atmosphere in the modules. These valves remained on the modules during shipment and storage. Isolation of the Intermediate Heat Exchangers began in January of 1968.

Following the isolation of the modules and prior to their removal from the vaults, bellows support braces and guide locking plates were installed on the module bellows on the shell side to retain the modules in a rigid position within their support frames, for removal and shipment.

An intermediate heat exchanger module with its support frame was removed from its position in the vault, Figure 38, by use of the reactor high bay crane and was held in an upright position while a shipping skid was attached to the side of the support frame; then the IHX module, support frame, and shipping skid were tilted down to the reactor high bay floor and prepared for shipment, Figure 39. The six Intermediate Heat Exchanger Modules were all shipped by truck to Idaho Nuclear Corp., NRTS, Central Facilities Area, Scoville, Idaho. The final shipment was made in June of 1968.

3-5-68

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Figure 37. HNPF Intermediate Heat Exchanger

Figure 38. IHX Module Being Removed from Vault

7709-4084

Figure 39. IHX Module With Shipping Skid Being Lowered to Floor

D. PUMP REMOVAL

1. Primary Sodium Pumps

The removal of the primary sodium pumps was performed in accordance with Retirement Activity Specification No. 5, Reaction of residual Sodium and Retirement of Primary Sodium Systems.

The three primary sodium pumps, Figure 40, were located north of the reactor and in a line parallel to the intermediate heat exchanger isolation structure.

To initiate the removal of the primary sodium pumps, the electrical services and instrumentation lines were disconnected. The oil seal pump and sodium piping were then disconnected from the pump. The pump motor and bearing seal assemblies were removed from the pump assemblies and stored in a temporary storage area above the IHX cells. The pumps were then removed from their outer casings and were transferred to the pump wash cell. After the pumps were removed from their outer casings, a gasketed steel plate was installed over the outer pump casings to provide a gas tight seal for the primary sodium system.

After the first wash cycle, each pump was removed from the wash cell for visual inspection. The pump diffusers were then removed, and the inner barrel and impeller were returned to the wash cell for a second washing. The pump assemblies were transferred to Pit 19 in the reactor high bay where the impellers were removed from the inner barrels. The inner barrels were then transferred to the wash cell for a third washing. The pump diffusers and impellers were also rewashed in the pump wash cell; and after washing, each part was hand washed with acetone for a final decontamination. The pump parts were then reassembled for shipment.

The primary sodium pump motor, bearing seal assemblies, and pump assemblies were crated and shipped to the Idaho Nuclear Corp., NRTS, Central Facilities Area, Scoville, Idaho for storage.

The outer pump casings remained in the primary sodium loops until the loops were steam cleaned. They were then removed from the primary sodium system and shipped to Scoville, Idaho for storage with the pumps.

PRIMARY PUMP:

IMPELLER DIAMETER: 59 in.
 BOWL DIAMETER: 109 in.
 BARREL DIAMETER: 67 in.
 LENGTH OF BARREL: 22 ft
 SUCTION NOZZLE: 36 in.
 DISCHARGE NOZZLE: 32 in.

7-6095-132-11

Figure 40. Free-Surface Sodium Pump

The primary pump balancing-legs disconnected from the primary sodium pumps during the pumps separation from the primary sodium system were considered to be scrap and were shipped to Beatty, Nevada for burial.

2. Secondary Sodium Pumps

The removal and shipment of the three secondary sodium pumps was performed in accordance with Retirement Activity Specification No. 6, Reaction of Residual Secondary Systems; and detailed procedure RDP-6-1, Isolation of the Secondary Sodium Service System and Disassembly of the System.

To initiate the removal of the secondary sodium pumps, the electrical services, instrumentation connections, and the oil seal pump were disconnected from the three secondary sodium pumps. The pump drive motors and bearing seal assemblies were then removed from the pump assemblies and placed in temporary storage on the reactor high bay floor on shipping skids designed and fabricated to support the pump drive motor assemblies until packaged for shipment. The pumps were then removed from their outer casing and transferred to the pump wash cell for cleaning. After washing, the pump assemblies were transferred to Pit 19 and disassembled; each pump component was individually washed in the wash cell. The secondary pumps were not reassembled after washing, but were packaged in separate crates for shipment.

The secondary sodium pumps were shipped to the Liquid Metals Engineering Center, Santa Susana, California on October 9, 1967.

The outer secondary pump cases were sealed and removed from the secondary sodium system according to detailed Procedure RDP-6-3, Isolate and Remove the Secondary Pump Cases from their Related Systems. The secondary pump cases were not washed. A gasketed steel plate was installed over the pump casing before it was removed from the secondary sodium system. The inlet and outlet pipes were sealed by welding a seal plate over the opening of the pipe stubs.

One secondary sodium pump casing was shipped to the Liquid Metals Engineering Center, Santa Susana, California. The remaining two secondary pump cases were shipped to Idaho Nuclear Corp., NRTS Central Facilities Area, Scoville, Idaho for storage.

3. Sodium Service EM Pumps

There were four electromagnetic pumps in the HNPF sodium service systems. One EM pump was used as a sodium transfer pump to transfer sodium from the sodium loading station to the sodium storage tanks, another of the EM pumps was in the secondary service system. The EM transfer and secondary-sodium pumps were not contaminated. These were removed from the sodium system during the removal of the secondary sodium system piping and were shipped to Pacific Northwest Labs, Richland, Washington in April of 1968.

The remaining two EM pumps were located in parallel in the primary sodium service system and were removed from that system during the removal of the primary sodium service system piping. The throat sections and connecting pipe stubs from the primary sodium EM pumps were contaminated with radioactive material. They were considered to be radioactive waste and were shipped to Beatty, Nevada for burial. The primary EM pump coils, cage, frames, and capacitors were packaged and shipped to the Liquid Metals Engineering Center, Santa Susana, California.

VIII. REFERENCES

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10. J. W. Wunderlich et al., "Potential Effects of Sodium Hydroxide on the HNPF Reactor Vessel - Hallam Retirement Program," AI-AEC-MEMO-12733 (August 9, 1968)

APPENDIX A
SPECIFICATIONS AND PROCEDURES

LIST OF ACTIVITY SPECIFICATIONS

<u>Spec. No.</u>	<u>Title</u>
1.	Primary Sodium Disposal
2.	Secondary Sodium Disposal
3.	Disposal of Irradiated Fuel
4.	Disposal of Unirradiated Fuel
5.	Reaction of Residual Primary Sodium and Retirement of Primary Sodium System
6.	Reaction of Residual Secondary Sodium and Retirement of Secondary Sodium System
7.	Disposal of Contaminated and Irradiated Material
8.	Reactor Isolation
9.	Retirement of Auxiliary Systems
10.	Securing of Isolation Structure
11.	Retirement of Radioactive Waste Facility
12.	Final Closeout of the Facility

LIST OF RETIREMENT DETAILED PROCEDURES

<u>Proc. No.</u>	<u>Title</u>
1-1	Isolation of the Primary Fill Tanks
1-2	Removal of Primary Sodium Fill Tanks and Transfer to Railroad Cars
2-1	Isolation of the Secondary Sodium Fill Tanks
2-2	Removal of Secondary Sodium Fill Tanks and Transfer to Railroad Cars
3-1	Disposal of Irradiated Fuel
3-2	Loading of the Six-Element Irradiated Fuel Shipping Cask at the HNPF
4-1	Disposal of Unirradiated Fuel
5-1	Isolation of Primary Sodium Service System and Disassembly of the System
5-2	Removal of the Primary Sodium Block and Throttle Valves and Isolation of the Reactor from Its Related Sodium Systems
5-3	Isolate Intermediate Heat Exchanger Modules from Their Related Primary System
5-4	Isolation and Removal of Primary Pump Balancing Legs
5-5	Reaction of Residual Sodium in Reactor
5-6	Reaction of Residual Sodium in the Primary Heat Transfer Loops
5-7	Isolation and Removal of Primary Pump Cases
6-1	Isolation of the Secondary Sodium Service System and Disassembly of the System
6-2	Isolate the Intermediate Heat Exchangers from Their Related Secondary Systems
6-3	Isolate and Remove the Secondary Pump Cases from Their Related Systems
6-4	Isolation of Steam Generator Evaporators and Superheaters from Secondary Sodium Systems
6-5	Removal of Secondary Heat Transfer Piping and Components
6-6	Disassembly and Removal of Steam Generators
8-1	Final Closure of Reactor and Containment Vessel Pipes
8-2	Final Closure of Reactor Top Shield and Reactor Structure Pipe Hangers
9-1	Dismantling and Removal of Feedwater and Steam System Piping and Components

APPENDIX B
ACTIVITY SPECIFICATION NO. 5 – REACTION OF RESIDUAL
PRIMARY SODIUM AND RETIREMENT OF
PRIMARY SODIUM SYSTEM

I. PURPOSE

The purpose of this activity specification is to provide for reaction or removal of residual sodium remaining in the primary sodium system and to provide for disposal of primary sodium system hardware and components.

II. ACTIVITY PLAN

For purposes of this activity specification, the primary sodium system will include the primary sodium main heat transfer system, the reactor vessel, and the primary sodium service system.

Sodium has been drained from the primary sodium system into primary fill tanks. The sodium filled primary fill tanks will be disposed of as per Activity Specification No. 1, "Primary Sodium Disposal." Residual sodium remains in certain portions of the system. Primary sodium system components and hardware will be classified in categories and disposed of accordingly. Component category classifications refer basically to the ultimate methods of disposal of the component. These category classifications are contained in Part II, C3, of the HNPF Retirement Plan and are summarized below:

Category A - Property authorized by CH for shipment off site except as scrap. Such property will be removed from its present location and shipped off site to AEC designated recipients.

Category B - Property which will be removed from its present location and shipped off site as scrap.

Category C - Property which will remain in its present location at the HNPF.

Category D - Property which will be removed from its present location and placed in adequately contained subsurface volumes.

Category E - Property which will remain in its present location but whose ownership will be transferred to CPPD.

Category F - Property which must be removed to accomplish the retirement of the HNPF and which CH authorizes to be removed from its present location pending determination of its ultimate disposition. Such property will be provided with suitable protection against damage and temporarily stored until its final status as Category A, B, or D material is decided.

The reactor vessel will be steam cleaned in place as described in Section IV and Appendix I of this activity specification. Other primary system components and hardware will be disposed of as follows:

Category A and F items will be removed from the system, residual sodium present will be removed and/or reacted as necessary, and prior to ultimate disposal the item will be sealed containing an inert atmosphere if necessary. Category B items will be removed from the system and residual sodium present removed or reacted as necessary. Residual sodium in Category C items will be reduced by reaction and/or physical removal to the specified limits for its safe containment. Where alternatives are stated, the steps employed shall be the most economical of those determined to be technically feasible.

III. ENGINEERING

A. Engineering, Scheduling, and Requirements

Operations connected with this activity will be scheduled in coordination with other HNPF Retirement activities. However, work contemplated under this activity is largely independent of other activities.

The methods of reaction of residual primary sodium and retirement of the primary sodium system were chosen as a result of system engineering analysis of the problems involved.

Major subcontractual work items, engineering studies, and equipment design and procurement to be conducted in support of this activity are listed below:

1. Design, procurement, and installation of sodium pumping apparatus for removing the residual sodium heel remaining in the reactor vessel.
2. Detailed design, equipment procurement, and system modifications to permit the in-place steam cleaning of the reactor vessel.
3. Detailed design, equipment procurement, and system modifications to permit the in-place reaction and/or removal of residual sodium in primary system components, other than the reactor vessel, as necessary (e.g., primary main heat transfer system piping).
4. Subcontractual work in connection with design, equipment procural, and existing hardware and structures modifications to permit the removal and/or reaction of residual sodium in Category A, B, and F items. It is recommended, based on engineering feasibility and economic considerations, that auxiliary sodium cleaning facilities be provided. The pump wash cell will be used for sodium cleaning operations which can be economically and reliably performed by remote controlled steam cleaning. Auxiliary facilities will be required for the majority of sodium cleaning operations (e.g., manual removal, swabbing, water flush, heated oil bath, and manual steam cleaning).
5. Heavy equipment removal subcontracts for such removal as is deemed necessary.
6. Miscellaneous engineering studies, design, and equipment procural in support of this activity.

B. Health and Safety

This activity will be carried out under normal HNPF and CPPD Safety regulations. Radiological safety will be in accord with the HNPF Operations Manual Volume Ia, Part III, and 10CFR20 regulations. Hardware lifting and moving will be conducted in accord with the CPPD Safety Manual.

C. Standards

Welding procedures, welding qualifications, and inspection for welding of all components utilized for reaction fluid containment during in-place steam cleaning operations shall conform to the ASME Boiler and Pressure Vessel Code, Section VIII.

Care will be exercised to minimize damage to Category A and F components during removal, handling, and cleaning operations. Category B and C components are scrap items and require no special handling to preserve the equipment.

Criteria for sodium removal and cleaning operations are contained in Part IV and Appendix I of this activity specification.

IV. METHOD OF ACCOMPLISHMENT AND SEQUENCE OF EVENTS

Sodium pumping apparatus shall be procured and utilized to pump the residual sodium heel from the reactor vessel. The depth of the residual sodium heel will be measured before and after pumping.

Primary sodium fill tanks will be removed and the primary sodium service system will be modified in accord with Activity Specification No. 1, "Primary Sodium Disposal."

Each component for which an approved CH purchase order has been received will be disposed of in accord with its respective removal requirement.

It should be noted that category designations, as described in Section II of this activity specification, define the ultimate disposal of the item.

A. Main Heat Transfer System

1. The residual sodium heel in the reactor vessel will be pumped out to reduce the quantity of residual sodium to a minimum. The average depth of the sodium heel after pumping shall be approximately 1/4-inch or less in depth.
2. The reactor vessel will be isolated from the primary system and steaming lines and hardware installed as described in Appendix I.
3. The primary balancing legs will be removed from the system and disposed of as Category C items.
4. The IHX units will be isolated and sealed by cutting and capping all primary sodium lines leading to the IHX units inside the IHX cells. Cutting and capping will be accomplished so that an inert atmosphere is maintained within the IHX. Secondary sodium lines will be cut and capped as per Activity Specification No. 6. Ultimate disposal of the IHX units will be as Category A items.
5. The primary main heat transfer system 14- and 16-inch piping will be modified to permit in-place steaming reaction of residual sodium. The modification essentially consists of providing steam inlet lines from the steam mixing tank (to be utilized in reactor vessel steaming) and exhaust lines to the condenser (to be utilized in reactor vessel steaming), isolating the main loops from the Na service system by cutting and capping and installing a bypass line to replace the flow path formerly provided by the IHX units. The primary pumps have already been removed from the system. The primary pump cases will be modified to permit in-place steam cleaning of the 14- and 16-inch main piping. Sodium piping of lesser diameter (6", 4", 3", and 2") that is not utilized for steaming activities will be removed from the primary pipe galleries and ultimately disposed of as Category B items.
6. The primary pumps, pump cases, and associated instrumentation and hardware will ultimately be disposed of as Category A items.
7. After modifications are complete the main primary piping (14- and 16-inch) will be steam cleaned in place one loop at a time utilizing essentially the same procedure as that to be employed in steaming the reactor vessel (Appendix I). The primary pump cases will ultimately be removed from the system and the 14- and 16-inch piping will be sealed with a dry nitrogen atmosphere and disposed of as Category C (left in place).

8. The reactor vessel will then be steamed as described in Appendix I.

B. Primary Sodium Service System

1. Primary sodium service system hardware, components, and piping for which approved CH purchase orders are not received will be classified as Category B items and will be removed from the system with the following exceptions:
  - a) Pipe sections which extend through vault walls and which are difficult to remove will be classified as Category C and left in place. Residual sodium present will be removed and/or reacted in the most expeditious and economic manner.
  - b) Piping which contains relatively insignificant amounts of sodium will be classified as Category C items and left in place or stored nearby in accord with operational convenience and economic considerations. For purposes of determining whether residual sodium is present in "significant" quantities the following procedure will be followed:
    - 1) The piping will be cut into convenient lengths so that internal surfaces may be visually inspected.
    - 2) If sodium is not present in layers exceeding 1/8-inch in thickness, the piping will be classified as Category C. Sodium layers of 1/8-inch or less will react with moisture from the atmosphere within approximately 30 days. This procedure will be followed provided that the radioactivity of the residual sodium present does not present a problem from the contamination control standpoint. If contamination control presents a problem, then the piping will be removed and disposed of as Category B material (with residual sodium being removed and/or reacted as necessary).
2. Residual sodium in the below-grade Category B primary sodium system items will be removed and/or reacted as necessary utilizing the pump wash cell and/or auxiliary cleaning facilities. If residual sodium removal in a Category B item is technically or economically unfeasible, the item will be suitably sealed and/or packaged and shipped off site for disposal elsewhere.

APPENDIX I
(of Appendix B)

Reaction of Residual Sodium in the Reactor Vessel

Residual sodium in the Hallam Nuclear Power Facility reactor vessel will be reacted with a nitrogen-steam mixture. Significant considerations are summarized below under two headings:

A. System Modifications and Preliminary Tasks and, B. Process Control and Instrumentation.

A. System Modifications and Preliminary Tasks

1. The residual sodium heel in the reactor vessel will be pumped out to reduce the quantity of residual sodium in the reactor vessel to a minimum. The average depth of the sodium heel after pumping shall be approximately 1/4-inch or less in depth.
2. The reactor vessel will be isolated from the main heat transfer system by cutting and capping the following lines on the primary pipeway side of the gallery seals:

Quantity	Size	Material Type	Remarks
3	14"	304	Na inlet
3	16"	304	Na outlet
1	6"	304	Moderator coolant inlet
3	3"	304	One fill and drain and two vents
3	18"	Carbon Steel	Guard pipe
3	20"	Carbon Steel	Guard pipe
1	10"	Carbon Steel	Guard pipe
1	6"	Carbon Steel	Guard pipe for fill and drain

3. A steam-nitrogen mixing station to introduce the reaction fluid into the reactor vessel and exhaust lines and a condensation tank to exhaust reaction products to the reactor building stack will be designed and installed.
4. A central control and instrumentation panel will be designed and installed for monitoring and control of the process.

B. Process Control and Instrumentation

1. General Description of Process

- a. The reactor vessel and associated piping will be preheated to 250°-300° utilizing existing procedures and equipment and heated nitrogen flow.
- b. Nitrogen-steam mixtures at 250°-300°F. will be introduced into the systems for reaction purposes. The initial steam/nitrogen ratio will be zero (i.e., pure heated nitrogen will be introduced into the system). The steam to nitrogen ratio will be slowly increased and the following parameters monitored.

- (1) Oxygen content
- (2) Hydrogen content
- (3) Pressure
- (4) Selected reactor vessel temperatures

- (5) Pressure
- (6) Vessel effluent temperature
- (7) Flow rates (both inlet and exit)

Instrumentation will be provided so that mass flows in and out of the reactor vessel may be estimated on the basis of recorded data.

- c. Steam flow rates, steam-nitrogen percentages, and reaction times to be utilized will be specified in the Detailed Procedures. These operational criteria will be based on experimental studies of the process conducted by Atomics International.

2. Normal Operating Conditions

Tentative values for the normal conditions under which the steaming process will be operated are listed below. These values are based on the studies of the process conducted by Atomics International. The Detailed Procedures will contain more definitive data for the process as necessary.

- a. Reactor vessel pressure, 1 to 6 psig.
- b. Reactor vessel atmosphere temperature, 250°-450°F.
- c. Reactor vessel structure temperature (bottom of vessel) 250°-550°F.
- d. Oxygen content of reactor vessel, trace quantities only.
- e. Nitrogen and steam mixture input to reactor vessel. 250°-300°F.

3. Tentative Maximum and Minimum Values

- a. Maximum deliberate reactor vessel operating pressure to be utilized for steaming operations, 10 psig
- b. Maximum reactor vessel pressure for flushing and venting operations (normal or emergency), 15 psig
- c. Maximum reactor vessel oxygen content, 1/2%
- d. Maximum vessel structure temperature (TC's on bottom of reactor vessel), 650°F
- e. Minimum temperature of reactor vessel structure, 240°F
- f. Maximum reactor vessel atmosphere temperature, 600°F
- g. Minimum reactor vessel pressure, 1/2 psig

The above maximums and minimums are to a greater or lesser extent arbitrary but are currently considered to be reasonable operational values based on studies conducted to date. The above parameters will be more precisely defined as necessary in the Detailed Procedures. It is assumed that the operator will take corrective action when significant deviations from normal values are observed before the above values are reached. Routine operation somewhat outside the above limits is not necessarily hazardous and may be permitted provided there is an operational reason for such procedures and appropriate approval is granted.

However, in no event will steaming operations be continued if 2% or greater oxygen concentrations are present since such oxygen concentrations would very likely indicate a gross malfunction of the system. Pressures in excess of 25 psig are considered to be possibly damaging to the reactor bellows (rupture of the bellows does not present a hazard

to personnel but would present other problems). Prolonged vessel atmosphere and structural temperatures in excess of 700°F. are considered to represent an excessively fast rate of Na-steam reaction due to the use of an unreasonably high steam flow rate and/or steam/nitrogen ratio for the amount of sodium present. Reactor temperature should be maintained at 240°F. or above to avoid any hazards due to possible steam condensation within the reactor vessel. Reactor pressure should be maintained at greater than atmospheric to preclude the possibility of air inleakage to the system.

4. Alarms

Tentative alarms and their setpoints are listed below. As discussed in Item 3 above, the setpoints are to a greater or lesser extent arbitrary and final values and/or additional alarms will be contained in the Detailed Procedures.

- a. High oxygen content in reactor vessel, 1/2%
- b. High reactor vessel pressure, 2 psig above desired operating pressure
- c. High reactor vessel atmosphere temperature, 550°F
- d. High reactor vessel structural temperature, 650°F
- e. Low reactor vessel structural temperature, 240°F
- f. Low reactor vessel pressure, 1/2 psig
- g. Automatic action high pressure alarm (4 psig above desired operating pressure)
- h. High radiation effluent alarm.

All alarms shall give both visible and audible alarm signals at the central control and instrumentation panel.

5. Operator Action in Case of Alarm

Detailed operator actions in case of alarm will be contained in the Detailed Procedures. The general nature of operator action in case of alarm is briefly summarized below.

Upon high vessel oxygen content alarm the operator shall cut off steam flow and commence nitrogen flushing operations until oxygen content returns to normal values.

Upon high reactor vessel pressure alarm, high reactor vessel atmosphere temperature alarm, and high reactor vessel structural temperature alarm the operator shall reduce steam and total flow rate as necessary to return conditions to normal. The automatic action high pressure alarm will automatically shut off both the steam and nitrogen supplies to the reactor vessel.

Upon low reactor vessel temperature alarm the operator shall energize the reactor vessel heaters as necessary to return conditions to normal.

Upon low reactor vessel pressure alarm the operator shall increase nitrogen flow rate as necessary to return conditions to normal.

Upon high radiation effluent alarm the operator shall begin to reduce steam and nitrogen flow rate preparatory to shutting down steaming operations and an investigation will be immediately initiated as to the cause of the alarm. Release of quantities of radioactive material sufficient to cause a high radiation effluent alarm is considered to be an extremely unlikely occurrence, and instrument malfunction and/or operations elsewhere in the plant

are considered to be more likely causes of the alarm. If investigation indicates the alarm is valid, the steaming process shall be shut down.

The occurrence of a valid high oxygen alarm, high radiation effluent alarm, or a significant sodium reaction pressure transient requires that, after immediate operator action is taken, the steaming operation will be shut down and an investigation of the causes of such occurrences will be conducted. Steaming operations may be recommenced after the observed phenomena have been explained, damage (if any) has been assessed, adequate provision made for prevention of future re-occurrence, and appropriate approval (to be specified in the Detailed Procedures) is obtained.

6. Reaction Completion

The reaction steaming process shall be performed in a manner such that there is a high degree of assurance, on the basis of studies conducted by Atomics International, that the reaction of residual sodium will be essentially complete. That is, reaction times utilized for specified steam percentages and flow rates shall be conservative in regard to the actual reaction times experimentally recorded to essentially completely react the estimated thicknesses of residual sodium present. These minimum reaction times, along with pertinent other data, will be specified in the Detailed Procedures. After the minimum reaction run times have been carried out, an additional requirement is that hydrogen evolution must be shown to be essentially negligible over a prolonged period (e. g., 6 hours) prior to the cessation of steaming operations. Finally, sodium samples of various thicknesses shall be placed in the reactor vessel prior to steaming operations and removed after the above requirements are met. Thus, a direct experimental check upon the efficacy of the steaming operations will be available.

After steaming operations have been completed, the steam supply will be cut off, the reactor vessel heaters will be energized as necessary, and heated nitrogen will be passed through the vessel to dry the NaOH which was formed during steaming. This will insure that the NaOH remaining in the vessel will be in solid form when the reactor is cooled to ambient temperature. Finally, the reactor vessel will be sealed with a dry nitrogen atmosphere such that a small positive gage pressure will be present under ambient conditions. The drying and nitrogen atmosphere sealing operation reduces dry corrosion rates to negligible proportions.

APPENDIX C
RETIREMENT DETAILED PROCEDURE 5-5
REACTION OF RESIDUAL SODIUM IN REACTOR

I. PURPOSE

This procedure establishes a step by step sequence for operating the steam-nitrogen process system for chemically reacting the residual sodium in the reactor vessel and its connecting line stubs.

II. DESCRIPTION

The activities to be accomplished by this procedure are as follows:

Establish specified pre-operating conditions on the steam-nitrogen system, the reactor vessel and the reactor vent system as designated by Section IV of this procedure. Admit the specified steam-nitrogen mixtures into the reactor vessel through the reactor line stubs as designated by Section IV of this procedure, determine that the residual sodium in the reactor has been chemically reacted as per specified tests of the gaseous effluent from the reactor and by inspection of the in-core sodium samples, terminate the steaming operation, purge inert gas through the system complex to insure that the moisture is removed, terminate the operation and permit the system, including the reactor, to cool to ambient temperature.

III. CRITERIA

A. Requirements and Prerequisites

1. This activity will be co-ordinated with other plant activities during the HNPF retirement program.
2. The reactor cavity atmosphere shall have $<2\text{-}1/2\%$ O_2 .
3. The steam-nitrogen system shall have been thoroughly tested, instruments calibrated and system operated to designed operating requirements per detailed procedure prior to initiating the operation detailed in Section IV of this procedure.
4. Sodium samples will be installed per Section VI of this procedure.
5. Drain tanks T-2, 3 and 5 will have a water heel deep enough to cover the end of the drain.
6. Pressures, temperatures, flow, composition of the steam-nitrogen mixture into the reactor vessel shall be recorded at 30 minute intervals during the initial reaction period (four hours); thereafter, this data shall be recorded at one-hour intervals.
7. Those personnel assigned to perform the work detailed in Section IV of this procedure shall be thoroughly checked out on all aspects of the operation. This requirement will include physically operating the system per detailed procedure prior to admitting process gases (steam-nitrogen) into the reactor.

B. Health and Safety

1. Special work permits are required for the work detailed in Section IV of this procedure.
2. A minimum of two operators shall be in attendance in the reactor building during the period that the steam-nitrogen system is in operation. A third operator will be required as necessary to record data in the control room.
3. Only authorized personnel will be permitted access to the primary pipe gallery during the steaming operation.

4. Personnel will wear protective clothing and equipment as deemed necessary by CPPD Health and Safety.
5. Activities that do not pertain to the sodium reaction operation will be conducted only by special CPPD Health and Safety permission on or near the reactor loading face or the sodium reaction process system exposed in the high bay area.

C. Standards

Work performed in connection with this activity will be carried out under normal HNPF Health and Safety regulations and in accordance with the specified requirements of this procedure.

IV METHOD OF ACCOMPLISHMENT AND SEQUENCE OF EVENTS (Refer to Dwg. N69012001, C Change)

1. All preparatory work shall be completed before starting the work of this section. Verify that V-5 is tagged open and the caps removed from the drains between V-15-15A, V-55-55A and V-57-57A.
2. Confirm that the LN_2 storage tank is full and the back-up N_2 supply is available and connected and the N_2 system is in normal operation with V-912 and V-46 open.
3. The reactor cover gas will be supplied at 1/2 psig through PCV-3 as indicated on PI-3 with V-51, V-52 block-reactor N_2 supply, and V-41 block-line 148-3-AA reactor drain line open. V-31, 67, 68 and 69 block valve to PI-2, 3, 4, and PRC-1 open, with the following valves closed:

V-1, 2, 3, 4, 6, 7, 11, 12, 15, 15A, 17, 18, 19, 24, 25, 26, 27, 28, 29, 30,
33, 36, 42, 43, 44, 45, 47, 53, 54, 55, 55A, 57, 57A, 58, 59, 60, 63, 65, 66
4. Energize the preheat on the gas mixing tank and all associated lines to and from the reactor. Check the preheat temperature on the instruments assigned and verify that the reactor and mixing tank temperature are 250° - 350°F. See date sheet IV-1.
5. Open V-58 and V-60 condenser cooling water and adjust flow on FI-608 to 140 gpm. Vent shell side by opening V-59 as required to remove any trapped gas in the condenser shell.

Initial Purge of System

6. Open the following valves:

V-63 throttle condenser gas outlet

V-33 inlet-hydrogen analyzer

V-17 block-reactor bypass

V-2 block N_2 supply

Verify that SV-3V is open and reactor pressure is above 1/4 psig. Verify N_2 system in normal operation (LN_2 system operating with 50 psig in the low pressure storage tank T-901), and energize N_2 heater H-2 with temperature control set at 150°F.

7. Open V-65, block N_2 sample to O_2 analyzer, and check out operation of O_2 analyzer. Verify N_2 supply <1/2% O_2 .
8. Check SV-2V open, Open V-7, and establish 20 cfm flow as read on FI-2. Compare FI-2 with FI-900. Throttle V-63 to maintain 1 psig minimum on PI-4.

9. Energize N₂ heater H-1 with temperature on TIC-1 set at 325°F.
10. Observe temperature on TRC-1 and pressure on PRC-1. Record all data on Data Sheet IV-1. TIC-1 should not drop below 250°F.
11. Observe O₂ analyzer and purge until O₂ is <1/2%.
12. Increase nitrogen flow on FI-2 to 150 cfm in 40 cfm increments using V-6 and V-7. Record all pressures and temperatures on Data Sheet IV-1. Adjust V-63 as required to maintain 1 psig minimum back pressure on PI-4 at all times.
13. Close V-6, V-7, and V-63. Install caps on the drains between V-15-15A and V-57-57A.
14. Close V-17 block-reactor bypass and open V-15 and V-15A block-reactor inlet header.
15. Open V-57 and V-57A block-reactor effluent outlet. Adjust V-6 and V-7 to establish 20 cfm flow through line 148, reactor fill and drain, as read on FI-2. Adjust V-63 to maintain 1 psig minimum on PI-4. Observe the O₂ analyzer; oxygen should remain below 1%. Temperature on TRC-1 should not go below 250°F. Record data on Data Sheet IV-1.
16. Increase N₂ flow to 150 cfm on FI-2 in 40 cfm increments using V-6 and V-7. Adjust V-63 to maintain 1 psig minimum on PRC-1. Record data on Data Sheet IV-1.
17. Purge the inlet and outlet headers by alternately opening and closing the following valves. Purge each combination for the indicated time. Observe that the O₂ analyzer remains <1/2% oxygen.

<u>Open</u>	<u>Close</u>	<u>Purge Time</u>
V-45	V-41	1 Min.
V-44	V-45	10 Min.
V-28	V-44	1 Min.
V-27	V-28	10 Min.
V-26	V-27	10 Min.
V-25	V-26	10 Min.
V-24	V-25	5 Min.
V-43	V-24	5 Min.
V-42	V-43	5 Min.

18. Open V-41, close V-42, close V-6 and V-7 and reduce N₂ flow on FI-2 to zero. Close V-63. Turn off the N₂ heater H-1.
19. Close V-15, V-15A, V-57 and V-57A. Remove the caps on the drains between the two pairs of valves.
20. Throttle V-5, open V-47, and warm up the steam header. Open V-1 block-steam supply V-11 mixing tank vent, and V-12 mixing tank drain. Crack open V-4 and warm up the steam header and mixing tank, limit pressure on PI-2 to 10 psig maximum and flow on FI-1 to 20 cfm maximum. Open V-66 and blow down FI-1, close V-66 and V-5.
21. Open V-29 block inlet laundry room hold-up tanks. Determine that V-1188 or V-1197 inlet to No. 1 or No. 2 hold-up tank is open.
22. Crack open V-7 and establish to 10 cfm N₂ flow on FI-2. Turn on N₂ heater H-1.
23. Open V-17 and then close V-11 and V-12 and adjust V-63 to hold 1 psig minimum on PI-4. Adjust N₂ flow to 100 cfm on FI-2 using V-6 and V-7. Adjust steam flow to 25 cfm on FI-1

using V-3 and V-4. Record all data on Data Sheet IV-1, -2. Determine that all systems are operating satisfactorily. Condensate flow should be observed in FI-5.

24. Verify that this procedure has been completed to step 23 and examine all data to determine if all conditions are satisfactory. Verify that item 5, Section III-A, "Requirements and Prerequisites," of this procedure has been completed and attach a signed-off training procedure. The system is now ready to begin steaming the reactor.

_____
Date

_____
Signature

Based on previous test experience, the following general information should be used as a guide for determining if the steam-sodium reaction is controlled and for making process adjustments for unusual circumstances.

- a. Reactor vessel gas temperature and pressure and the hydrogen evolution rate are the main data points which indicate the status of the reaction. The reaction is assumed to be controlled if the reactor vessel gas temperature and pressure remain relatively constant. However, if during steady-state operation reactor vessel gas temperature reaches 450°F or reactor vessel pressure reaches 6 psig, the steam flow should be immediately reduced in an attempt to correct the situation. When normal conditions are again established for a period of 30 minutes, the steam flow should be slowly returned to the operating point.
- b. During periods of changing steam-nitrogen gas mixture ratios it is possible that the above-noted unusual temperature and pressure conditions may be encountered. If this is the case, the nitrogen flow should be increased or the steam flow reduced (depending upon which flow is being adjusted) to the previous set point where essentially a steady-state condition existed. When conditions are again normal for a minimum of 30 minutes, attempts should be made to continue with the procedure.
- c. If at any time the reaction appears to be uncontrollable by simply reducing steam flow, the steam and nitrogen flow should be completely stopped and supervision notified immediately. Reactor cover gas will be maintained at 1/2 psig through PCV-3.

Caution

The following operating limits should not be exceeded:

PRC-1 Reactor pressure minimum 1/2 psig
maximum 6 psig

Reactor Vessel Atmosphere Effluent Temperature

TRC-1 Minimum 250°F
Maximum 450°F

Reactor Vessel (bottom) Temperature

Minimum 250°F
Maximum 550°F

Reactor cavity liner temperature maximum 225°F.

Oxygen content of reactor effluent and nitrogen supply <1/2%.

In the event that any of these conditions are approached or exceeded, refer to Section V, "Operator Action in Case of Alarm."

Steaming Reactor

25. Reduce steam flow to zero by closing V-3 and V-4 and crack open V-12 mixing tank drain. Nitrogen will bleed from the tank. Purge at 100 cfm N_2 for 5 minutes to dry out the system.
26. Close V-41, crack V-45, and observe N_2 flow into drain tank T-5. Replace the caps on the drain between V-15-15A, V-55-55A and V-57-57A.
27. Open V-57 and V-57A block-reactor effluent outlet. Open V-15 and V-15A block-reactor inlet header. Purge through V-45. Open 41, close V-17 and V-45.
28. Establish 100 cfm N_2 flow through the reactor. Adjust V-6 and V-7 as required. Adjust V-63 to hold 1 psig minimum back pressure on the reactor.
29. Crack open V-4 and slowly increase the steam flow on FI-1 to 25 cfm. Blow down the mixing tank and close V-12 when dry steam appears. Monitor the hydrogen analyzer and observe for an increase in hydrogen. Record all data at 30 minute intervals. Watch carefully the temperature on TRC-1 and monitor reactor vessel temperature. Compare with data taken in step 23. An increase in loading face shield and upper cavity liner temperatures is expected until these surfaces approach the temperature of the steam going into the reactor. The reactor effluent temperature TRC-1 is not expected to be appreciably higher than the inlet temperatures on TR-1 and TR-2. Notify supervisor if TRC-1 goes above 350°F. Temperatures will be limited to a maximum 450°F for normal operation.
30. After conditions have reached a quasi-steady condition, while holding steam flow constant at 25 cfm reduce nitrogen flow from 100 cfm to 60 cfm in 20 cfm increments waiting 10 minutes between changes. Increase steam flow from 25 cfm to 60 cfm in 17 cfm increments waiting 10 minutes between changes. Continue at a steaming rate of 60 cfm steam 60 cfm nitrogen, 50% mixture steam and nitrogen for 6 hours. Record all data at 30 minute intervals for 4 hours. Here after all data will be recorded at one-hour intervals.
31. Alternately increase nitrogen flow and then steam flow in 20 cfm increments at 10 minute intervals until flow is increased to a total of 300 cfm (150 cfm nitrogen and 150 cfm steam). Continue this steaming rate for at least 6 hours.

NOTE: The hydrogen detector will indicate at what rate the sodium-steam reaction is taking place. The hydrogen detector should be used as a guide for knowing when to proceed if the concentration of hydrogen is high enough to give a good indication on the detector.

32. Open V-44 and begin steaming 239-14-AA, close V-41. Steam for one hour or until the hydrogen detector shows a peak and then a decrease.
33. Open V-28 and blow down the header until dry steam is observed at the condensate drain into T-3 drain tank. TR-1 should read the same as TR-2 and not less than 250°F. Close V-28.
34. Open V-26 and introduce steam through line 237-14-AA. Throttle V-44 but leave enough flow to keep the line warm. Steam for one hour or until the hydrogen detector peaks and then shows a decrease.
35. Open V-27 and begin steaming line 238-14-AA, close V-26 and steam for one hour or until the hydrogen detector shows a peak and then a decrease.

36. Open V-24 and begin steaming line 231-16-AA, close V-27. Steam for one hour or until the hydrogen detector shows a peak and then a decrease.
37. Open V-25 and begin steaming line 284-6AA, close V-24 and steam for one hour or until the hydrogen detector shows a peak and then a decrease.
38. Open V-42 and begin steaming line 232-16-AA, close V-25. Steam for one hour or until the hydrogen detector shows a peak and then a decrease.
39. Open V-43 and begin steaming line 233-16 AA, close V-42. Steam for one hour or until the hydrogen detector shows a peak and then a decrease.
40. Open V-41, 42, 44, 24, 25, 26 and 27. Introduce steam through all lines into the reactor. Reduce nitrogen flow from 150 cfm to 60 cfm in 20 cfm increments waiting 10 minutes between changes. Increase steam from 150 cfm to 240 cfm in 20 cfm increments waiting 10 minutes between changes. Continue under 80% steam concentration for 10 hours.
41. Close V-42, 43, 44, 24, 25, 26 and 27. Steam through V-41 and line 148-3 for 1 hour.
42. Open V-44 and close V-41 and steam through line 239-14-AA for 1 hour.
43. Open V-28 and blow down the header until dry steam is observed at T-3 drain tank. Close V-28.
44. Open V-26 and steam through line 237-14-AA for 1 hour. Throttle V-44 but leave enough flow to keep the line warm.
45. Open V-27, close V-26 and steam line 238-14-AA for 1 hour.
46. Open V-24 and close V-27. Steam line 231-16-AA for 1 hour.
47. Open V-25 and close V-24. Steam line 284-6-AA for 1 hour.
48. Open V-42 and close V-25. Steam line 232-16-AA for 1 hour.
49. Open V-43 and close V-42. Steam line 233-16-AA for 1 hour.
50. Open V-41, 42, 44, 24, 25, 26 and 27. Monitor the hydrogen detector and continue steaming at 80% steam concentrations (60 cfm N₂, 240 cfm steam) until hydrogen has been undetected for 6 hours.
51. Reduce the steam and nitrogen flow to zero by closing V-3, 4, 6 and 7 and turn off N₂ heater H-1. Reactor pressure will be maintained at 1/2 psig through PCV-3. Close V-63, V-47, and V-1. Tag open V-5.
52. Remove one sodium sample from the reactor to determine if the sodium has all been reacted by the steam. If reaction of the first sample is not complete, repeat step 50. If the first sample is complete, remove the second sample to determine if it has been completely reacted. If both samples are done, proceed to the next step in this procedure. If the reaction is not complete on the second sample, repeat step 50. Repeat this process until examination of two samples indicates the reaction has been completed. All samples will be removed and handled according to Section VI of this procedure.
53. Prepare to purge the reactor. Set all preheat control points up to 450°F. Reset TAH-1 to 450°F. Close V-24, 42, and 43.
54. Turn on heater H-1, open V-6 and V-7 and purge the reactor with dry nitrogen holding nitrogen temperature outlet on both heaters as high as possible, not to exceed 450°F on

H-1 and 150°F on H-2, at a flow rate of 80 cfm minimum. Adjust V-63 to hold 1/2 psig back pressure on the reactor. Continue the purge for 24 hours.

55. Open V-24, 42 and 43. Continue the purge until the level control LC-1 has stopped collecting water.
56. Secure the purge by closing V-6, 7 and 63. Reactor pressure will be maintained at 1/2 psig through PCV-3. Secure nitrogen heaters H-1 and H-2.
57. Remove remaining sodium samples from the reactor according to Section VI of this procedure and verify that the samples are solid at room temperature.
58. Secure all reactor and reaction system heaters and allow the reactor to cool to ambient temperature. Maintain 1/2 psig on the reactor through PCV-3 or use an alternate source of N₂ supply if reaction system is disconnected from the reactor before the reactor reaches ambient temperature.

V. CORRECTIVE ACTION IN CASE OF ALARM

1. Reactor high pressure (alarm set at 8 psig, automatic action closes SV-IV and SV-2V at 10 psig; manual reset is required to reopen SV-1 and SV-2). Open V-63, verify flow in FI-4 and determine that total flow on FI-1 and FI-2 has not exceeded 300 cfm. Close V-3 and V-4, reduce steam flow as required to reduce pressure in the reactor. Determine that PCV-3 is operating satisfactorily.
2. Reactor low pressure (alarm set at 1/4 psig). Automatic action closes SV-3V when alarm sounds. SV-3V opens when alarm clears. Check to see that PCV-3 is operating and PI-3 is indicating pressure. Throttle V-63 to increase reactor pressure to 1 psig as indicated on PI-3.
3. High oxygen (alarm set at 1/2% O₂ in nitrogen). Determine that the reactor is above atmospheric pressure and that the nitrogen supply has not been contaminated with oxygen. A high alarm might be expected during initial purging operations. Only trace quantities of oxygen should be present during the steaming operation. If a high oxygen > 1/2% occurs during the steaming operation, secure the steam flow by closing V-3 and V-4 and continue the hot nitrogen purge until the trouble can be located and corrected.
4. Reactor atmosphere effluent high temperature (alarm set at 550°F). Reduce the steam and nitrogen flow being careful to maintain the same steam-nitrogen flow ratio. This will reduce the reaction rate and effluent temperature should decrease.
5. Reactor atmosphere effluent low temperature (operating limit 250°F). A low effluent temperature on TRC-1 indicates that something is wrong with the temperature of the steam and/or nitrogen supply. Determine that the steam supply is 350°F as read on TI-1 and the nitrogen supply is 300°F or higher as read on TIC-1. Discontinue steaming if the temperature on TRC-1 goes below 225°F until the cause of the low temperature can be determined and corrected.
6. Reactor vessel bottom low temperature. Alarm set at 250°F as read on TI-21-26 through 30 on main board in control room. Determine that reactor bottom heaters are on and operating.

7709-5401

Figure 1. Sodium Sample Suspended from Shield Plug

7. Reactor vessel bottom high temperature alarm set at 550°F as read on TI-21-26 through 30. Determine that the bottom heaters are off. If necessary, reduce steam and nitrogen flow to slow the reaction rate and reduce reactor vessel bottom temperatures.
8. Reactor cavity liner and loading face shield temperatures. There is no alarm but the cavity liner and loading face shield temperature as read on TI-18 on the main board should be limited to a maximum of 225°F.

VI. PREPARATION, INSTALLATION AND REMOVAL OF SODIUM SAMPLES USED TO DETERMINE COMPLETION OF THE REACTION PROCESS IN THE REACTOR

Each sample will consist of 2 cups, one suspended 3 inches above the other. The bottom cup will contain 12 grams of sodium; the upper cup will contain 25 grams. The sample cups will be fabricated from a section of 2 inch diameter schedule 40 stainless steel pipe 1-1/2 inches long with a flat plate welded ~ 1/4" deep on the bottom. The sample cup assembly will be suspended 1 foot below the loading face shield as shown in Figure 1.

Six samples will be installed in the reactor by attaching them to the bottom of a reactor shield plug and inserting them into the reactor using the short gas lock. A temporary operating procedure will be developed for this operation at the time the work is to be done. Sample locations are shown in Figure 2.

The samples will be removed, using the same procedure that covers the installation. The samples will be removed as required to monitor the reaction and drying process.

7709-5402

Figure 2. Sodium Sample Locations

VALVES

Legend

- V-1 Block - Steam Supply
- 2 Block - N₂ Supply
- 3 Throttle - Steam Supply - Mixing Tank T-1
- 4 Throttle - Steam Supply - Mixing Tank T-1
- 5 Drain - Condensate - Steam Supply Line
- 6 Throttle - N₂ Supply - Mixing Tank T-1
- 7 Throttle - N₂ Supply - Mixing Tank T-1
- 8 Nonreturn - Steam Inlet - Mixing Tank T-1
- 9 Nonreturn - N₂ Inlet - Mixing Tank T-1
- 10 Nonreturn - Mixing Tank Vent
- 11 Vent - Mixing Tank T-1
- 12 Drain - Mixing Tank Blowdown
- 13 Nonreturn - Mixing Tank Drain
- 14 Nonreturn - Mixing Tank Outlet
- 15 Block - Reactor Inlet Header
- 16 Nonreturn - Reactor N₂ Supply
- 17 Block - Reactor Bypass
- 18 Bypass - PCV-3 Reactor N₂ Supply
- 19 Drain - Condensate Drain Tank T-2
- 20 Nonreturn - Line 231-16-AA Reactor Outlet
- 21 Nonreturn - Line 284-6 AA Moderator Coolant
- 22 Nonreturn - Line 237-14-AA Reactor Inlet
- 23 Nonreturn - Line 238-14-AA Reactor Inlet
- 24 Block - Line 231-16-AA Reactor Outlet
- 25 Block - Line 284-6-AA Moderator Coolant
- 26 Block - Line 237-14-AA Reactor Outlet
- 27 Block - Line 238-14-AA Reactor Outlet
- 28 Drain - Condensate From Line 13-6-MI
- 29 Block - Inlet Laundry Room Hold Up Tanks
- 30 Block - Inlet Decontamination Room Hold Up Tanks
- 31 Block - PI-4
- 32 Nonreturn - Condensate Line to T-3
- 33 Block - Inlet - Hydrogen Analyzer
- 34 Check - Drain Line for Steam Supply Line to T-1
- 35 Check - Drain Line for Steam Supply Line to T-1
- 36 Drain - Condensate Drain Tank T-3
- 37 Nonreturn - Line 148-3-AA Reactor Drain Line
- 38 Nonreturn - Line 232-16-AA Reactor Outlet
- 39 Nonreturn - Line 233-16-AA Reactor Outlet
- 40 Nonreturn - Line 239-14-AA Reactor Outlet
- 41 Block - Line 148-3-AA Reactor Drain Line
- 42 Block - Line 232-16-AA Reactor Outlet
- 43 Block - Line 233-16-AA Reactor Outlet
- 44 Block - Line 239-14-AA Reactor Outlet
- 45 Drain - Condensate From Line 23-6-MI
- 46 Block - Nitrogen System
- 47 Block - Steam System
- 48 LC-2 - Drain
- 49 Nonreturn - Condensate Line to T-5
- 51 Inlet - Pressure Control Valve PCV-3
- 52 Block - Reactor N₂ Supply
- 53 Drain - Condensate Drain Tank T-5
- 54 Block Reactor N₂ Vent
- 55 Block - N₂ Purge Reactor Bypass
- 56 Nonreturn - Reactor Outlet (Reactor Effluent)
- 57 Block - Reactor Effluent Outlet
- 58 Block - Condenser Cooling H₂O Inlet
- 59 Vent - Condenser Water Side
- 60 Block - Condenser Cooling H₂O Outlet
- 61 Nonreturn - Vent - O₂ Analyzer
- 62 Nonreturn - Vent - Hydrogen Analyzer

- V-63 Throttle - Condenser - Gas Outlet
- 64 Nonreturn - Condenser - Gas Outlet
- 65 Block - N₂ Sample to O₂ Analyzer
- 66 Blowdown - FI-1 Steam Flowmeter
- 67 Block - PI-2
- 68 Block - PI-3
- 69 Block - PRC-1
- 70 Heater H-2 Bypass
- 71 Heater H-2 Block
- 72 Heater H-2 Block
- 73 PCV-1 Block
- 74 PCV-1 Block
- 75 PCV-1 Bypass

- PCV-1 Nitrogen Supply from GN₂ Trailer Set at 80 psig
- 2 N₂ Sample to O₂ Analyzer Set at 1/2%
- 3 N₂ Cover Gass to Reactor Set at 1/2 psi

- SV-1V Steam Supply High Pressure Reactor Cut Off
- 2V N₂ Supply High Pressure Reactor Cut Off
- 3V Reactor Vent Low Pressure Cut Off
- 4V Control Valve Condenser Drain

- PSV-1 Relief Valve - Mixing Tank - Set at 25 psig
- 2 Relief Valve - 1/2 lb. Reactor Supply Set at 10 psig

APPENDIX D-

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APPENDIX D-1
DERIVATION OF EQUATIONS FOR ANALYSIS
OF SODIUM-STEAM REACTION

DATA:

(t_d) = time at which data point was taken based on twenty-four hour clock
(hr and min)

(T_n) = nitrogen temperature upstream of inlet orifice (TI-6) (°F)

(ΔP_n) = nitrogen orifice pressure difference (FI-2) (%)

(T_s) = steam temperature upstream of inlet orifice (TI-1) (°F)

(ΔP_s) = steam orifice pressure difference (FI-1) (%)

(P_b) = barometric pressure (in. Hg)

(P_m) = mixing tank pressure (PI-2) (psig)

(X_h) = hydrogen fraction in condenser gas effluent (%)

(P_a) = atmospheric pressure (psia)

$$P_a = 0.491 P_b \quad \dots(1)$$

(P_t) = mixing tank pressure in psia

$$P_t = P_a + P_m \quad \dots(2)$$

(M_s) = steam flow rate through inlet orifice (lb-mole/min)

Diagram of Steam Inlet Piping

From N. C. Ebaugh "Engineering Thermodynamics," Lancaster Press, 1937
p. 115, Equation 13

$$M = 8.02 C_d F A_2 \frac{P_1}{\sqrt{R T_1}} \sqrt{\frac{K}{K-1}} \sqrt{\left(\frac{P_2}{P_1}\right)^{(K+1)/K} - \left(\frac{P_2}{P_1}\right)^{2/K}} \quad \text{lb/sec} \quad \dots(3)$$

p. 114, Equation 10

$$F = \frac{1}{\sqrt{1 - \left(\frac{D_2}{D_1}\right)^4 \left(\frac{P_2}{P_1}\right)^{2/K}}} \quad \dots(4)$$

where

C_d = orifice discharge coefficient (assumed = 0.61)

A_2 = orifice area (= 1.248 in.²)

FORTTRAN
Symbol

NT

TI6

FI2

TI-1

FI1

PB

PI2

RH2

PA

PT

P_2 = orifice downstream pressure (psia with A_2 in sq in.)

P_1 = orifice upstream pressure (psia with A_2 in sq in.)

K = ratio of const. pres. spec. ht. to const. vol. spec. ht. (= 1.29 for steam under operating conditions)

R = gas constant (= 85.4 ft-lb/lb-°F for steam under operating conditions)

T_1 = orifice upstream temperature (°F abs)

D_2 = ID of orifice approach pipe (= 2.067 in. for 2 in. - Sch. 40 pipe)

D_1 = orifice diameter (= 1.260 in.)

Equations 3 and 4 may be combined and rearranged to the form

$$M = 8.02 C_d A_2 P_1 \left\{ \frac{\left(\frac{K}{K-1} \right) \left(1 - r^{(K-1)/K} \right)}{R T_1 \left[\left(\frac{1}{r} \right)^{2/K} - \beta^4 \right]} \right\}^{1/2} \quad \dots(5)$$

where

r = orifice pressure ratio (= P_2/P_1)

β = diameter ratio (= D_2/D_1)

Direct measurement of P_2 was not taken during the operation. Measurement of the gage pressure at P_1 was taken during part of the operation using PI-13. Analysis is of the data indicated reasonably accurate correlation between the flow-meter reading (FI-1) and the pressure difference [(PI-13) - (PI-2) = $P_1 - P_t$] to be given by the equation: (see Appendix D-2)

$$(P_1 - P_t) = 0.0733 (FI-1) \quad \dots(6)$$

For the 33 points examined, the average deviation between measured and calculated values was 0.253 psi, the standard (root mean square) deviation was 0.336 psi. Considering that the minimum reading for (PI-2) was 9.8 psig and assuming a standard atmosphere of 14.0 psia (at 1440 ft-6 in. elevation) the average deviation in calculating P_1 would be about 1.02%, the standard deviation about 1.36%.

Using Equation 6, P_1 is calculated from:

$$P_1 = P_t + 0.0733 (FI-1) \quad \dots(7)$$

The orifice meter (FI-1) measured the pressure difference across the orifice as a fraction of the rated pressure difference (= 100 in. H_2O). The read-out scale on this meter was a square root scale ranging from 0 to 100% such that 100% is equivalent to 100 in. H_2O . The pressure difference (ΔP_g) is therefore

$$\Delta P_g = [(FI-1)/100]^2 \text{ in. } H_2O \quad \dots(8)$$

$$\therefore r = 1 - 0.000361 (FI-1)^2 / P_1 \quad \dots(9)$$

where 1 in. H_2O = 0.0361 psi.

P

R

For the orifice flow in lb-mole/min

$$M_s = \frac{60}{18}(M) \text{ lb-mole/min}$$

where

60 = seconds/min

18 = molecular weight of steam

Using the calculated results from Equations 7 and 9 and putting numerical values into Equation 5 we have

$$M_s = \frac{60}{18}(8.02)(0.61)(1.248)P_1 \left\{ \frac{\left(\frac{1.29}{1.29-1} \right) (1-r)^{(1.29-1)/1.29}}{85.4T_1 \left[\left(\frac{1}{r} \right)^{2/1.29} - \left(\frac{1.260}{2.067} \right)^4 \right]} \right\}^{1/2} \quad \dots(10)$$

$$= 20.35P_1 \left\{ \frac{4.45(1-r)^{0.225}}{85.4T_1 \left[\left(\frac{1}{r} \right)^{1.55} - 0.138 \right]} \right\}^{1/2} \text{ lb/mole/min} \quad \dots(11)$$

where $T_1 = 460 + (T_s)$

$$M_s = 4.64P_1 \left\{ \frac{(1-r)^{0.255}}{(460 + T_s) \left[\left(\frac{1}{r} \right)^{1.55} - 0.138 \right]} \right\}^{1/2} \text{ lb-mole/min} \quad \dots(12)$$

XST

(M_n) Nitrogen flow rate through inlet orifice in lb-mole/min

The nitrogen flow rate is calculated in the same manner as the steam flow rate, using Equation 5 and the following constants:

$C_d = 0.61$ (assumed)	$D_2 = 1.114$ in.
$A_2 = 0.974$ in. ²	$D_1 = 2.067$ in.
$K = 1.40$	Molecular wt = 28
$R = 55.2$ ft-lb/lb-°F	

The upstream pressure (P_1) for the nitrogen orifice is based on analysis of the data taken on pressure gage (PI-14) which was also used during part of the operation. Analysis of the data indicated correlation between the flowmeter reading (FI-2) and the pressure difference [(PI-14) - (PI-2) = $P_1 - P_t$] to be given by the equation:

$$(P_1 - P_t) = 0.0465(\text{FI-2}) \text{ psia} \quad (\text{see Appendix D-2}) \quad \dots(13)$$

For the 36 points examined, the average deviation between measured and calculated values was 0.164 psi, the standard (root mean square) deviation was 0.217 psi.

Considering that the minimum value for $(P_1 - P_t)$ was 3.4 psig and assuming a standard atmosphere of 14.0 psia (at 1440 ft-6 in. elevation) the average deviation in calculating P_1 would be about 0.94%, the standard deviation about 1.19%.

Using Equation 6, P_1 is calculated from:

$$P_1 = P_t + 0.0465(FI-2) \quad \dots(14)$$

P

The nitrogen orifice meter (FI-2) was similar to the steam orifice meter (FI-1), having a 0 to 100% square root scale with 100% equivalent to 100 in. H_2O . The pressure drop (ΔP_n) is therefore

$$\Delta P_n = [(FI-2)/100]^2 \text{ in. } H_2O \quad \dots(15)$$

$$\therefore r = 1 - 0.000361(FI-2)^2/P_1 \quad \dots(16)$$

R

(NOTE: The symbols P_1 , r , and T_1 are used for both nitrogen and steam calculations since these are not stored simultaneously in the computer.)

Using the calculated results from Equations 14 and 16 and putting numerical values into Equation 5 we have

$$M_n = \frac{60}{28} (8.02)(0.61)(0.974)P_1 \left\{ \frac{\left(\frac{1.40}{1.40-1} \right) (1 - r^{(1.40-1)/1.40})}{55.2T_1 \left[\left(\frac{1}{r} \right)^{2/1.40} - 0.085 \right]} \right\}^{1/2} \quad \dots(17)$$

$$M_n = 10.2P_1 \left\{ \frac{3.5(1 - r^{0.286})}{55.2T_1 \left[\left(\frac{1}{r} \right)^{1.428} - 0.085 \right]} \right\}^{1/2} \text{ lb-mole/min} \quad \dots(18)$$

where $T_1 = 460 + T_n$

$$M_n = 2.57P_1 \left\{ \frac{(1 - r^{0.286})}{(460 + T_n) \left[\left(\frac{1}{r} \right)^{1.428} - 0.085 \right]} \right\}^{1/2} \text{ lb-mole/min} \quad \dots(19)$$

XN2

(W_s) Steam flow rate through inlet orifice in lb/hr

$$W_s = 18(60)M_s \quad \dots(20)$$

$$W_s = 1080M_s \quad \dots(21)$$

HST

(Q_n) Nitrogen flow through inlet orifice in scfm

$$Q_n = \frac{R' T_{std}}{P_{std}} M_n \text{ scfm} \quad \dots(22)$$

where

R' = universal gas const. (= 1545 ft-lb/lb-mole-°F)

T_{std} = standard temperature (= 520°F abs)

P_{std} = standard pressure (= 2116.8 psf)

$$Q_n = 379 M_n \quad \dots(23)$$

HN2

(t_m) Data time in minutes of 1440 minute day

The data time was recorded in hours (a) and minutes (b) based on a 24-hour clock. The form of the record is

$$(100a) \text{ hours} + (b) \text{ minutes}$$

Therefore, in terms of a 1440 minute clock, the time is:

$$t_m = 60 \frac{(100a)}{100} + b$$

In the computer, this was performed by integer calculations which truncate the remainder on division

$$t_m = \left[\frac{(100a + b)}{100} \right] 60 + (100a + b) - \left[\frac{(100a + b)}{100} \right] \text{ minutes} \quad \dots(24)$$

MDT

where $\left[\frac{(100a + b)}{100} \right] = a$

(Δt) Time interval between data points in minutes

The time interval is calculated from the difference of the recorded times of successive data points

$$\Delta t = t_{m(i+1)} - t_{m(i)} \text{ minutes} \quad \dots(25)$$

XMTI

where subscript (i) is the number of the data point

($S_{s(i)}$) Steam flow over time interval in lb-moles

It is assumed that the average steam flow over the time interval is equal to the average of the flow rates at the beginning and end of the interval (data points) multiplied by the time interval

$$S_{s(i)} = \frac{M_{s(i)} + M_{s(i+1)}}{2} (\Delta t) \quad \dots(26)$$

STI

(C_{s(j)}) Total steam flow since start of operation in lb-mole

$$C_{s(j)} = \sum_{i=1}^{j} S_{s(i)} \quad \dots(27) \quad \text{CUMST}$$

where limit j = number of data point at end of last time interval considered.

(F_s) Volume fraction of steam in gas entering reactor in %

Based on Mole % = Volume %

$$F_s = \frac{M_s}{M_s + M_n} 100 \quad \dots(28) \quad \text{FST}$$

(M_o) Sodium reaction rate (with steam) in lb-mole/min

Assuming steady state conditions (i.e., nitrogen inlet flow = nitrogen outlet flow) and the condenser effluent gas is all nitrogen and hydrogen. The lb-mole/min of hydrogen leaving the condenser (M_h) is:

$$M_h = \frac{(X_h/100)}{1 - (X_h/100)} M_n \quad \text{lb-mole/min} \quad \dots(29)$$

The chemical equation for the steam-sodium reaction indicates that one mole of hydrogen is formed when two moles of sodium react with steam

Therefore for each mole of hydrogen in the effluent gas, two moles of sodium were reacted in the system

$$M_o = 2M_h$$

$$M_o = \frac{(X_h/50)}{1 - (X_h/100)} M_n \quad \text{lb-mole/min} \quad \dots(30) \quad \text{XNA}$$

(W_o) Sodium reaction rate in lb/hr

$$W_o = 23(60)M_o$$

$$W_o = 1380M_o \quad \text{lb/hr} \quad \dots(31) \quad \text{HNA}$$

(S_{o(i)}) Sodium reacted in time interval in lb-mole

$$S_{o(i)} = \frac{M_{o(i)} + M_{o(i+1)}}{2} \Delta t \quad \text{lb-mole} \quad \dots(32) \quad \text{XNAI}$$

($C_{o(j)}$) Total amount of sodium reacted since start of operation in lb-mole

$$C_{o(j)} = \sum_{i=1}^{i=j} S_{o(i)} \quad \text{lb-mole} \quad \dots(33) \quad \text{CUMNA}$$

($e_{(i)}$) Process efficiency during time interval in percent

For 100% efficiency each lb-mole of steam entering the reactor would combine with one lb-mole of sodium. The process efficiency over the interval is defined as the ratio of rate of sodium reacted (in lb-mole/min) to rate of steam entering the reactor vessel (in lb-mole/min)

$$e_{(i)} = \frac{M_o}{M_s} 100 \quad \text{percent} \quad \dots(34) \quad \text{EFI}$$

($\eta_{(j)}$) Average process efficiency since start of operation in percent

$$\eta_{(j)} = \frac{C_{o(j)}}{C_{s(j)}} 100 \quad \text{percent} \quad \dots(35) \quad \text{EFA}$$

($D_{s(j)}$) Total steam flow since start of operation in lb

$$D_{s(j)} = 18C_{s(j)} \quad \dots(36) \quad \text{CST}$$

($D_{o(j)}$) Total amount of sodium reacted since start of operation in lb

$$D_{o(j)} = 23C_{o(j)} \quad \dots(37) \quad \text{CNA}$$

ANALYSIS OF DATA FOR CORRELATION BETWEEN INLET ORIFICE METER
READINGS (FI-1 AND FI-2) AND PRESSURE DROP BETWEEN
UPSTREAM ORIFICE PRESSURE TAPS AND MIXING TANK

ANALYSIS OF DATA FOR CORRELATION BETWEEN INLET ORIFICE METER READINGS
(FI-1 AND FI-2) AND PRESSURE DROP BETWEEN UPSTREAM ORIFICE PRESSURE TAPS
AND MIXING TANK

STEAM INLET

(M) Number of points examined

$$N = 33$$

($\sum \Delta P_1$) Sum of measured pressure drops (psi)

$$\sum \Delta P_1 = [(PI-13) - (PI-2)] = 84.1$$

($\sum FI-1$) Sum of orifice meter readings (%)

$$\sum (FI-1) = 1147.4$$

The assumed form of the correlating equation is:

$$\Delta P = m(FI-1)$$

(m) Average proportionality constant

$$m = \sum \Delta P_1 / \sum (FI-1) = 84.1 / 1147.4$$

$$m = 0.0733$$

($\sum e_1$) Sum of absolute differences between measured pressure drops and calculated pressure drops

$$\sum e_1 = \sum |\Delta P_1 - m(FI-1)| = 8.34$$

($\bar{e}_1$) Average deviation

$$\bar{e}_1 = (\sum e_1) / M = 8.34 / 33 = 0.253$$

($\sum e_1^2$) Sum of squares of difference between calculated and measured values

$$\sum e_1^2 = \sum [\Delta P_1 - m(FI-1)]^2 = 3.720$$

(σ_1) Standard deviation

$$\sigma_1 = \left[(\sum e_1^2 / M) \right]^{1/2} = 3.720 / 33^{1/2}$$

$$\sigma_1 = 0.336$$

NITROGEN INLET

(N) Number of points examined

$$N = 36$$

($\sum \Delta P$) Sum of measured pressure drops

$$\sum \Delta P_2 = \sum [(PI-14) - (PI-2)] = 70.5$$

($\sum FI-1$) Sum of orifice meter readings

$$\sum (FI-2) = 1518.9$$

The assumed form of the correlating equation is:

$$\Delta P_2 = n(FI-2)$$

(n) Average proportionality constant

$$n = \sum \Delta P_2 / \sum (FI-2) = 70.5 / 1518.9$$

$$n = 0.0465$$

($\sum e_2$) Sum of absolute differences between measured pressure drops and calculated pressure drops

$$\sum e_2 = \sum |\Delta P_2 - n(FI-2)| = 5.91$$

($\bar{e}_2$) Average deviation

$$\bar{e}_2 = (\sum e_2) / N = 5.91 / 36 = 0.164 \text{ psi}$$

($\sum e_2^2$) Sum of squares of difference between calculated and measured values

$$\sum e_2^2 = \sum [\Delta P_2 - n(FI-2)]^2 = 1.698$$

(σ_2) Standard deviation

$$\sigma_2 = \left[(\sum e_2^2) / N \right]^{1/2} = 1.698 / 36$$

$$\sigma_2 = 0.217 \text{ psi}$$

RECOMMENDED EQUATIONS FOR ESTIMATING PRESSURE DIFFERENCE BETWEEN ORIFICE UPSTREAM PRESSURE TAPS AND MIXING TANK

Steam:

$$\Delta P_1 = 0.0733(FI-1) \text{ psi}$$

Nitrogen:

$$\Delta P_2 = 0.0465(FI-2) \text{ psi}$$

	C	HNPf RETIREMENT	00000100
	C	CALCULATION OF RESULTS FROM STEAMING OF REACTOR VESSEL	00000200
	C		00000300
	C	INITIAL VALUE SET	00000400
		IMPLICIT REAL*8(A-H,G-Z)	00000450
ISN 0002		MTE=0	00000500
ISN 0003		YST=0.0	00000600
ISN 0004		CUMST=C.0	00000700
ISN 0005		YN2=0.0	00000800
ISN 0006		CUMN2=0.0	00000900
ISN 0007		YNA=0.0	00001000
ISN 0008		CUMNA=C.0	00001100
ISN 0009		MS=735	00001200
ISN 0010		N=C	00001210
ISN 0011		5 M=C	00001220
ISN 0012		10 N=N+1	00001225
ISN 0013		IF(N.GT.20) GO TO 75	00001226
ISN 0014		WRITE(6,11)	00001230
ISN 0016		11 FORMAT('1'/' ',4X,'HNPf RETIREMENT'/' ',4X,'RESULTS FROM STEAMING	00001232
ISN 0017		10f REACTOR VESSEL'/'0',T9,'1',T16,'2',T29,'3', T39,'4	00001234
		2',T48,'5',T58,'6',T67,'7',T77,'8',T89,'9',T98,'10' /	00001236
		3' ',T6,'ELAPSED',T14,'STEAM', T25,'NITROGEN', T36,'VOLUME',	00001238
		4T44,'TOTAL', T55,'VOLUME', T63,'SODIUM', T74,'PROCESS',	00001240
		5T85,'SODIUM', T96,'TOTAL', /' ',T6,'TIME', T14,'FLOW',T25,'FLOW',	00001242
		6T36,'PERCENT', T44,'STEAM', T55,'PERCENT', T63,'REACTION',	00001244
		7T74,'OPER.', T85,'REACTED', T96,'PROCESS' /' ', T14,'RATE',	00001246
		8T25,'RATE',T36,'STM.',T44,'INLET',T55,'HYD.',T63,'RATE',T74,'EFF.'	00001248
		9,T96,'EFF.')	00001250
ISN 0018		WRITE(6,12)	00001252
ISN 0019		12 FORMAT('0',T6,'HR/MIN', T14,'LB/HR',T25,'SCFM',T44,'LB.',	00001254
		1T63,'LB/HR',T74,'PERCENT',T85,'LB.',T96,'PERCENT' /)	00001256
		READ INPUT DATA	00001400
ISN 0020	C	15 READ(5,20) NT,TI6,FI2,TI1,FI1,PB ,PI2,RH2	00001500
	C		00001600
ISN 0021	C	20 FORMAT(15,7D7.1)	00001700
	C		00001800
	C	START CALCULATIONS	00001900
ISN 0022		M=M+1	00002000
	C	ATMOSPHERIC PRESSURE PSIA	00002100
ISN 0023		PA= 0.491*PB	00002200
	C	MIXING TANK PRESSURE PSIA	00002300

ISN 0024		PT=PA+PI2	00002400
	C	STEAM INLET FLOW RATE LB-MOLE/MIN	00002500
ISN 0025		IF(FI1.LE.0.01)GO TO 30	00002600
ISN 0027		P=PT+0.0733*FI1	00002601
ISN 0028		R=1.0-0.000361*(FI1**2)/P	00002602
ISN 0029		XST=4.64*P*SQRT((1.0-(R**0.255))/((460.0+T11)*((1.0/R)**1.55 - 0.1	00002603
		138)))	00002604
ISN 0030		GO TO 31	00002605
ISN 0031	30	XST=0.0	00002606
ISN 0032	31	CONTINUE	00002607
	C	STEAM FLOW RATE (LB/HR)	00002610
ISN 0033		HST=XST*1080.0	00002620
	C	NITROGEN INLET FLOW RATE LB-MOLE/MIN	00002700
ISN 0034		IF(FI2.LE.0.01)GO TO 40	00002800
ISN 0036		P=PT+0.0465*FI2	00002801
ISN 0037		R=1.0-0.000361*(FI2**2)/P	00002802
ISN 0038		XN2=2.57*P*SQRT((1.0-(R**0.285))/((460.0+T16)*((1.0/R)**1.428 - 0.0	00002803
		185)))	00002804
ISN 0039		GO TO 41	00002805
ISN 0040	40	XN2=0.0	00002806
ISN 0041	41	CONTINUE	00002807
	C	NITROGEN FLOW RATE (SCFM/60F)	00002810
ISN 0042		HN2=XN2*379.0	00002820
	C	DATA TIME (MINUTES OF 1440 MINUTE DAY)	00002900
ISN 0043		NHR=NT/100	00003000
ISN 0044		MDT=NHR*60	00003100
ISN 0045		MDT=MDT+NT-100*NHR	00003200
	C	TIME INTERVAL MINUTES	00003300
ISN 0046		IF(MDT.LT.MS) GO TO 50	00003400
ISN 0048		MTI=MDT-MS	00003500
ISN 0049		MS=MDT	00003600
ISN 0050		GO TO 55	00003700
ISN 0051	50	MDT=MDT+1440	00003800
ISN 0052		IF(MDT.LT.MS)GO TO 71	00003850
ISN 0054		MTI=MDT-MS	00003900
ISN 0055		MS=MDT-1440	00004000
ISN 0056	55	CONTINUE	00004100
	C	ELAPSED TIME FROM START MINUTES	00004200
ISN 0057		MTE=MTE+MTI	00004300
	C	ELAPSED TIME FROM START (HR/MIN)	00004310
ISN 0058		MHR=MTE/60	00004320
ISN 0059		MIN=MTE-60*MHR	00004330

	C	STEAM MASS FLOW OVER TIME INTERVAL LB-MOLE	00004400
ISN 0060		XMTI=MTI	00004500
ISN 0061		STI=(YST+XST)*XMTI/2.0	00004600
ISN 0062		YST=XST	00004700
	C	ACCUM. STM. FLOW FROM START LB-MOLE	00004800
ISN 0063		CUMST=CUMST+STI	00004900
	C	NITROGEN MASS FLOW OVER TIME INTERVAL LB-MOLE	00005000
ISN 0064		XNI=(YN2+XN2)*XMTI/2.0	00005100
ISN 0065		YN2=XN2	00005200
	C	ACCUM. N2 FLOW OVER TIME INTERVAL LB-MOLE	00005300
ISN 0066		CUMN2=CUMN2+XNI	00005400
	C	INLET STEAM FRACTION PERCENT	00005500
ISN 0067		FST=XST*100.0/(XST+XN2)	00005600
	C	SODIUM REACTION RATE LB-MOLE/MIN	00005700
ISN 0068		XNA=RH2*XN2/((1.0-RH2/100.)*50.0)	00005800
	C	SODIUM REACTION RATE (LB/HR)	00005810
ISN 0069		HNA=XNA*1380.0	00005820
	C	SODIUM REACTED IN TIME INTERVAL LB-MOLE	00005900
ISN 0070		XNAI=(YNA+XNA)*XMTI/2.0	00006000
ISN 0071		YNA=XNA	00006100
	C	ACCUM. NA REACTED IN TIME INTERVAL LB-MOLE	00006200
ISN 0072		CUMNA=CUMNA+XNAI	00006300
	C	PROCESS EFFICIENCY DURING TIME INTERVAL PERCENT	00006400
ISN 0073		IF(FI1.LE. 0.0001) GO TO 60	00006410
ISN 0075		EFI=100.0*XNA/XST	00006500
ISN 0076		GO TO 65	00006510
ISN 0077	60	EFI=0.0000	00006520
ISN 0078	65	CONTINUE	00006530
	C	ACCUM. PROCESS EFFICIENCY PERCENT	00006600
ISN 0079		IF(MTI.LE. 1) GO TO 66	00006610
ISN 0081		EFA=100.0*CUMNA/CUMST	00006700
ISN 0082		GO TO 68	00006710
ISN 0083	66	EFA=0.000	00006720
ISN 0084	68	CONTINUE	00006730
	C	ACCUM. STM. FLOW IN (LB)	00006800
ISN 0085		CST=CUMST*18.0	00006900
	C	ACCUM. N2 FLOW IN (LB)	00007000
ISN 0086		CN2=CUMN2*28.0	00007100
	C	ACCUM. NA REACTED (LB)	00007200
ISN 0087		CNA=CUMNA*23.0	00007300
	C		00007400
ISN 0088		WRITE(6,70)PHR,MIN,HST,HN2,FST,CST,RH2,HNA,EFI,CNA,EFA	00007500

ISN 0089	70	FORMAT(' ',T4,I4,T8,'/',T9,I3,T13,0PF8.3,T24,F7.3,T35,F7.2,T44,F8.	00007600
		11,T55, PF7.3,T63,F7.3,T73,F7.3,T84,F7.3,T96,F7.3)	00007700
ISN 0090		IF(M.LT.32)GO TO 15	00007800
	C	LINE COUNTER RESET	00007900
ISN 0092		GO TO 5	00008000
ISN C093	71	WRITE(6,72)	00008010
ISN 0094	72	FORMAT(' ',4X,'TIME ERROR')	00008020
ISN 0095	75	CONTINUE	00008100
ISN C096		CALL EXIT	00008150
ISN C097		STOP	00008175
ISN C098		END	00008200

KEYPUNCH INPUT SHEET
DATA

29972-1

T. J. BEARDMAN

71/731/131

2395

7967

Page 1 of 1

SODSTMI

Program No. & Title																Card Stock				Char. Set	
NT (15)	TIG (F7.1)	FIZ (F7.1)	TII (F7.1)	FII (F7.1)	PB (F7.2)	PIZ (F7.1)	RH2 (F7.2)														
1	4	8	12	16	20	24	28	32	36	40	44	48	52	56	60	64	68	72	76	80	
1	1215	305.0	56.0	235.0	12.0	28.46	10.0	2.50											10001		
2	1243	301.0	57.8	223.0	5.0	28.46	8.1	3.60											10002		
3	1315	297.0	56.0	225.0	10.0	28.46	9.5	3.05											10003		
4	1345	282.0	64.3	228.0	10.0	28.45	11.2	3.20											10004		
5	1415	272.0	64.4	229.0	10.0	28.45	11.1	3.00											10005		
6	1445	270.0	64.3	228.0	10.0	28.45	11.1	2.80											10006		
7	1515	282.0	56.0	225.0	10.0	28.45	9.4	2.90											10007		
8	1545	297.0	56.0	225.0	10.0	28.45	9.4	3.00											10008		
9	1615	298.0	56.0	225.0	10.0	28.46	9.4	2.90											10009		
10	1645	300.0	49.0	220.0	10.0	28.46	7.4	2.90											10010		
11	1715	332.0	40.0	217.0	10.0	28.46	6.0	3.40											10011		
12	1745	345.0	45.0	218.0	10.0	28.47	6.6	3.70											10012		
13	1815	335.0	45.2	218.0	10.0	28.47	6.8	3.75											10013		
14	1845	330.0	45.0	218.0	10.0	28.48	6.5	3.70											10014		
15	1915	331.0	45.0	218.0	10.0	28.49	6.5	3.33											10015		
16	1945	335.0	41.0	218.0	10.0	28.50	6.4	3.10											10016		
17	2015	330.0	41.0	217.0	10.0	28.51	6.2	3.26											10017		
18	2045	328.0	40.0	218.0	12.5	28.52	6.5	3.18											10018		
19	2115	328.0	40.0	220.0	15.0	28.53	7.0	3.27											10019		
20	2145	329.0	40.0	220.0	15.0	28.54	7.0	3.50											10020		
21	2215	330.0	40.5	217.0	11.0	28.56	6.1	3.65											10021		
1	4	8	12	16	20	24	28	32	36	40	44	48	52	56	60	64	68	72	76	80	

FORM 114-B-50 NEW 1-67

LISTING OF INPUT DATA FOR CALCULATION OF RESULTS
FROM SODIUM-STEAM REACTION

AI-AEC-MEMO-12736

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Appendix D-4

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Program No. & Title

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KEYPUNCH INPUT WORKSHEET
DATA

299724

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SFDSTM1

Program No. & Title

Card Stock

Char Set

	1	4	8	12	16	20	24	28	32	36	40	44	48	52	56	60	64	68	72	76	80
1	1245	255.0	63.0	259.0	35.0	28.67	20.0	0.65												10043	
2	1345	254.0	62.5	257.0	37.0	28.64	20.5	0.55												10044	
3	1445	255.0	61.5	260.0	38.0	28.62	20.8	0.40												10045	
4	1545	256.0	61.5	260.0	37.8	28.60	20.8	0.30												10046	
5	1645	256.0	62.2	261.0	38.0	28.58	20.5	0.20												10047	
6	1745	255.0	66.8	260.0	39.8	28.54	17.6	0.19												10048	
7	1845	250.0	70.0	258.0	40.0	28.56	16.1	0.09												10049	
8	1945	245.0	70.0	260.0	42.0	28.57	16.1	0.15												10050	
9	2045	244.0	70.0	260.0	42.0	28.59	15.9	0.05												10051	
10	2145	243.0	71.0	259.0	42.0	28.60	15.8	0.10												10052	
11	2245	229.0	67.0	261.0	42.0	28.60	18.6	0.05												10053	
12	2345	228.0	68.0	262.0	40.0	28.60	18.0	0.05												10054	
13	0045	230.0	70.0	262.0	40.0	28.61	12.6	0.12												10055	
14	0145	279.0	45.0	255.0	42.0	28.61	8.5	0.10												10056	
15	0245	323.0	36.0	258.0	54.0	28.60	9.5	0.40												10057	
16	0345	331.0	33.0	285.0	59.0	28.60	12.3	0.80												10058	
17	0445	345.0	33.0	300.0	74.0	28.59	13.9	0.82												10059	
18	0545	339.0	43.0	300.0	0.0	28.59	4.0	0.50												10060	
19	0645	339.0	43.0	300.0	0.0	28.60	3.6	0.25												10061	
20	0745	350.0	43.0	207.0	0.0	28.62	3.6	0.00												10062	
21	0845	343.0	36.0	244.0	34.0	28.62	6.9	0.00												10063	
	1	4	8	12	16	20	24	28	32	36	40	44	48	52	56	60	64	68	72	76	80

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Appendix D-4

KEYPUNCH INPUT WORKSHEET
DATA

29972-1

T J BOARDMAN

71/73/131

2395

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SPDSTMI

Program No. & Title																			Card Stock		Char. Ser.	
	1	4	8	12	16	20	24	28	32	36	40	44	48	52	56	60	64	68	72	76	80	
1	0945		337.0		36.0		284.0		34.0		28.64		6.9		0.02						10064	
2	1045		337.0		35.5		285.0		34.0		28.64		6.6		0.10						10065	
3	1145		336.0		34.0		295.0		48.5		28.64		8.6		0.11						10066	
4	1245		290.0		35.1		307.0		58.0		28.62		10.9		0.11						10067	
5	1345		296.0		27.1		304.0		57.0		28.59		9.1		0.10						10068	
6	1445		283.0		25.0		317.0		57.6		28.57		9.1		0.16						10069	
7	1545		287.0		25.0		317.0		58.0		28.54		9.4		0.08						10070	
8	1645		280 0		25.0		305.0		58.0		28.53		9.4		0.08						10071	
9	1745		270 0		25.0		303.0		58.0		28.50		9.5		0.06						10072	
10	1845		273 0		20.0		298.0		49.8		28.50		7.5		0.05						10073	
11	1945		278.0		20.0		294.0		50.0		28.48		7.4		0.05						10074	
12	2045		278.0		20.0		300.0		50.0		28.48		7.4		0.05						10075	
13	2145		278.0		20.0		303.0		50.0		28.48		7.3		0.05						10076	
14	2245		278.0		27.0		293.0		55.0		28.48		9.0		0.05						10077	
15	2345		283.0		24.0		292.0		47.0		28.47		13.4		0.00						10078	
16	0045		285.0		25.0		272.0		50.0		28.46		13.0		0.00						10079	
17	0145		283.0		25.0		303.0		53.0		28.46		11.2		0.00						10080	
18	0245		283 0		25.0		294.0		53.0		28.44		10.5		0.05						10081	
19	0345		283.0		24.0		294.0		52.0		28.44		10.8		0.02						10082	
20	0445		283.0		25.0		292.0		51.0		28.41		10.0		0.00						10083	
21	0545		282.0		24.0		294.0		50.0		28.40		11.6		0.00						10084	

KEYPUNCH INPUT ~~XXXXXXXX~~
DATA

29972-1 T.J. FORTNII
Deck Number Programmer

71/25/12/1
Dist/Dept/Group

2295
Phone

Date

7967
Job or Charge No.

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Program No. & Title

Card Stack

Char Set

	1	4	8	12	16	20	24	28	32	36	40	44	48	52	56	60	64	68	72	76	80
1	0645	283.0	24.0	295.0	51.0	28.40	12.0	0.00													1.0085
2	0745	289.0	25.0	299.0	57.0	28.40	9.1	0.24													1.0086
3	0845	291.0	25.0	305.0	58.0	28.37	9.9	0.04													1.0087
4	0945	282.0	25.0	312.0	58.2	28.37	9.9	0.17													1.0088
5	1045	282.0	25.0	305.0	58.1	28.36	9.9	0.15													1.0089
6	1145	281.0	24.0	306.0	58.1	28.35	9.9	0.10													1.0090
7	1245	282.0	25.0	305.0	58.1	28.31	9.9	0.05													1.0091
8	1345	281.0	24.5	303.0	58.0	28.21	9.9	0.03													1.0092
9	1445	280.0	25.0	315.0	58.0	28.26	9.9	0.01													1.0093
10	1545	280.0	24.5	310.0	58.1	28.22	10.0	0.01													1.0094
11	1645	278.0	24.8	300.0	58.0	28.18	9.9	0.00													1.0095
12																					
13																					
14																					
15																					
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17																					
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20																					
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