MASTER

75,000 KILOWATTS OF ELECTRICITY
BY NUCLEAR FISSION
AT THE
HALLAM NUCLEAR POWER FACILITY

By

F. C. GRONEMEYER, Atomics International
Project Manager, Hallam Nuclear Power Facility

and

J. W. MERRYMAN, Bechtel Corporation
Manager, Nuclear Plant Engineering

Paper to be presented at ASCE Convention
in Reno, Nevada on Thursday, June 23, 1960

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

A. Makes any warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

932 001

RELEASE APPROVED — AUTHORIZATION ON
FILE IN RECEIVING SECTION.

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

75,000 KILOWATTS OF ELECTRICITY
BY NUCLEAR FISSION
AT THE
HALLAM NUCLEAR POWER FACILITY

By

F. C. GRONEMEYER, *Atoms International*
Project Manager, Hallam Nuclear Power Facility

and

J. W. MERRYMAN, *Bechtel Corporation*
Manager, Nuclear Plant Engineering

Paper to be presented at ASCE Convention
in Reno, Nevada on Thursday, June 23, 1960

ACKNOWLEDGMENT

The assistance of the following persons is gratefully acknowledged for providing information contained in this paper.

R. C. Gerber, Chief Project Engineer, HNPF, Atomics International

H. A. Ross-Clunis, Project Engineer, HNPF, Atomics International

J. E. Mahlmeister, Project Engineer, HNPF, Atomics International

J. F. Stolz, Project Engineer, HNPF, Atomics International

John Gentles, Project Engineer, HNPF, Bechtel Corporation

CONTENTS

	Page
History of Project	1
Program Participants	4
Site Description	5
Component Development Program	6
Fuel Elements	6
Moderator - Reflector Assemblies	7
Control - Safety Rods	7
Mechanical Pumps	8
System Preheating	8
Sodium Instrumentation	9
Valves	9
Steam Generator	10
Cold Traps	10
Reactor Building	13
Building Ventilation	14
Fire Protection Systems	25
Reactor Structure	25
Loading Face Shield	26
Reactor Vessel and Internals	29
Containment Vessel	29
Thermal Shields and Thermal Insulations	31
Support Structure for Vessels, Thermal Shields, and Insulation	32
Cavity Liner and Reactor Cavity Diaphragm Seal	32
Reactor Core	35
Fuel Elements	35
Moderator and Reflector Elements	38
Reflector Filler and Dummy Elements	38
Reactor Design Data	39
Sodium Systems	43
Instrumentation and Control	44

CONTENTS

	Page
Sodium Instrumentation System	46
Reactor Plant Protective System	47
Radiation Detection and Monitoring	48
Fuel and Component Handling	51
Fuel and Moderator Handling Machine	52
Wash Cells	52
Storage Cells	52
Pick-Up Cell	53
Maintenance Cell	53
Shipping Casks	53
Portable Purge Unit	54
Auxiliary Systems	55
Special Design Features	58
Plant Control System	59
Advantages of Sodium Graphite Reactor Systems	62

FIGURES

	Page
1. Hallam Plant	viii
2. Current Stage of External Completion	2
3. Reactor Building	12
4. Reactor Building Layout Plan above Elevation of 1460 ft	15
5. Reactor Building Layout Plan below Elevation of 1460 ft	17
6. Reactor Building Layout Plan below Elevation of 1440 ft	19
7. Vertical Sections through Reactor Building	21
8. Current Stage of Completion within the Reactor Building	23
9. Cutaway View of the Reactor Core	27
10. Reactor Vessel Nearing Completion	28
11. Grid Plate with 141 Moderator Element Supports Nearing Completion	30
12. Reactor Cavity Liner	33
13. Fuel Element Assembly	36
14. Moderator-Reflector Element	37
15. Sodium Heat Transfer Circuit	42
16. Fuel Handling Machine	50

Figure 1. Hallam Plant

HISTORY OF PROJECT

The first power plant version of the Sodium Graphite Reactor (SGR) concept is being built for the United States Atomic Energy Commission and will be operated by the Consumers Public Power District of Nebraska at their Sheldon Station site in Hallam, near Lincoln, Nebraska.

This plant, called the Hallam Nuclear Power Facility (HNPF), will produce 75,000 net electrical kilowatts, using 800 psig, 825°F steam. The reactor will use uranium-molybdenum (U-10/Mo) as the initial fuel, graphite as moderator, and sodium as the heat transfer fluid. The plant will operate with higher reactor coolant temperatures, higher fuel element surface temperatures, and higher steam pressure and temperature than any other reactor currently authorized in the First Round of the USAEC Demonstration Power Reactor Program. The purpose of the Consumers SGR Project, as stated in the contract between the AEC and Consumers, is "to demonstrate the economic and technical practicability of central station power utilizing a sodium graphite nuclear reactor, by operating the plant for the primary purpose of producing a maximum amount of electrical energy therefrom...."

The design, construction, safety features, and operating procedures of the HNPF are based on experience obtained by operation of the Sodium Reactor Experiment (SRE), located north of Los Angeles in the Santa Susana Mountains, which was designed and constructed by Atomics International and operated for the AEC since April 1957.

Following the signing of a contract between the AEC and Consumers in September 1957, the current program was initiated in November 1957. The first of two prime objectives called for accomplishing the preliminary engineering in an effort to provide, by September 1958, a complete set of criteria for final design of the plant. The cost estimate and preliminary design were submitted to the AEC in September 1958. The second objective required completion of a prototype component development and test program to provide data for the final design of all critical components. Final engineering was initiated in October 1958 and virtually completed in June 1960. Site construction, started in April 1959, is presently approximately 33 percent complete.

Figure 2. Current Stage of External Completion

The schedule for the project provides for construction completion of the nuclear facility in January 1962, and for full power operation of the nuclear power plant in June 1962. To meet their power commitments, Consumers plans to place in operation a coal-fired plant in the spring of 1961. The nuclear facility will supply steam to the turbine which is initially installed with the coal-fired plant; subsequently a second turbine-generator will be added for the conventional boiler. Included in the scope of the nuclear facility are the sodium graphite reactor, the sodium heat transfer systems, the sodium steam generators, the overall power-plant control system for nuclear operation, the nuclear fuel-handling system, and various plant auxiliaries

PROGRAM PARTICIPANTS

Separate prime contracts for the Hallam Nuclear Power Facility exist between the Atomic Energy Commission and Atomics International, Bechtel Corporation, Peter Kiewit Sons', Inc., and Consumers Public Power District.

Atomics International, under its prime contract, is responsible for performing research and development, establishing and coordinating design criteria, preparing and coordinating preliminary design (except for the Administration Building), performing final design of major reactor systems and components, furnishing components designed by AI, providing installation instructions for items installed by AI, installing some of the components furnished by AI, and initial testing and operation of the Nuclear Facility.

Consumers Public Power District, under a prime contract with the AEC, is responsible for completing the preliminary design for the control room area, cooling towers, and steam generator services equipment; performing final design and providing installation instructions and specifications for steam generator services, administration building, and control room; furnishing the steam generator services equipment and the site; and operating the completed and accepted HNPF.

Bechtel Corporation, under its prime contract with the AEC, is responsible for independently preparing the design criteria and the preliminary design for the administration building and will cooperate with AI and CPPD in the completion of the preliminary design for the Nuclear Plant, performing final design and preparing construction drawings for the reactor building, piping and instrumentation, process system, radioactive waste facility, and electrical systems, and providing installation instructions for items to be installed by Peter Kiewit Sons', Inc.

Peter Kiewit Sons', Inc., under prime contract with the AEC, is responsible for furnishing and installing all components and systems in the reactor building which are not furnished by AI or CPPD, and for constructing the reactor building, substructure concrete foundations, vaults, superstructure, process systems, radioactive waste facility, and the administration building.

SITE DESCRIPTION

The site for the Hallam Nuclear Power Facility is located in Lancaster County in southeastern Nebraska. The facility is near the center of Section 19 in Township 7 North, Range 6 East.

The Hallam area of southeast Nebraska is located in Zone 1 of the Uniform Building Code map of seismic probability. Zone 1 is defined as an area where infrequent seismic disturbances may be expected to cause only minor damage

The reactor site is on relatively high ground on the northeast slope of the divide separating the watersheds of the Big Blue River, Salt Creek, and the north fork of the Nemaha River.

The climate in the Hallam area is subject to a wide seasonal range in temperature. The average rainfall at the site is 27.51 in./yr.

In the general area of the site, the prevailing winds are from the northwest during the winter months, and from the southeast during the summer months.

COMPONENT DEVELOPMENT PROGRAM

Due to the very considerable scale-up in power output and physical dimensions of the Consumers SGR reactor core and plant components over the SRE, and the desirability of incorporating improved materials and designs, the USAEC has supported a broad component development program. A major objective of this program is to confirm the suitability of each major component for its intended application before construction of the reactor plant. This program includes testing of both sub-components and full-scale prototypes under simulated reactor operating conditions. The component development program covers fuel elements, control and safety rods, control instruments, moderator and reflector assemblies, and sodium instrumentation, pumps, valves, and other sodium system components. It also includes the development, by suitable in-pile tests, of improved fuel and control poison materials. Hydraulic studies on models will be made to provide design data for the fuel elements and core structural components. Experimental determination of temperature gradients and stresses will be made in components subjected to high thermal stresses. Handling techniques for removal of all reactor core components are being developed with a full-scale mockup. All of the information derived from the above work was scheduled for completion early enough in the overall program to support the final design and the construction effort. During the construction period on the plant, proof tests in sodium at design temperatures and pressures will be run on all production units of such components as mechanical pumps and control and safety rods before they are installed in the plant. In addition, it is planned to run life tests on spare production units of these components.

Given below is a summary of the development work currently planned and in progress on major components:

FUEL ELEMENTS

Consumers SGR fuel elements will operate at higher surface temperatures than those for any other large-scale civilian power reactor in the present AEC Demonstration Program.

Atomics International has, through the development program, established the enrichment and quantity of the fuel for the first core loading. In addition to the studies for the first core, long-range refueling studies of the Hallam reactor fueled with uranium carbide have been made. The fuel rods are designed to accommodate more than the expected U-Mo swelling for a peak burnup of 15,000 Mwd/MTU at a maximum fuel temperature of 1250°F. Extrapolation of the available data indicates that the volume increase of the fuel (under HNPF conditions) will not exceed 14%. This corresponds to a sodium annulus of 0.020 in. between the fuel cladding and the fuel slugs. A 0.025-in. sodium annulus will be provided in the final design.

Each fuel rod consists of the uranium-molybdenum fuel material in a column, 13-1/2 feet high by 0.59 inch in diameter, enclosed in a thin-walled, stainless-steel jacket. The jacket is 0.640 inch in interior diameter, and 0.010 inch in wall thickness.

MODERATOR - REFLECTOR ASSEMBLIES

In addition to the increase in size of the graphite moderator units, 0.016-in.-thick Type 304 stainless-steel cladding was used, rather than the 0.035-in.-thick zirconium cladding used in the SRE. A fabrication development program was completed which involved forming of the required shapes to close dimensional tolerances and joining by heliarc welding in accordance with extremely high quality requirements. A number of full-scale prototypes were made for a test program aimed at determining the adequacy of design with respect to mechanical and thermal stress. In the full-scale tests, a group of seven moderator units were thermally cycled in a large sodium tank. In addition to these full-scale tests, other tests were run to prove out some of the detailed design features.

CONTROL - SAFETY RODS

The development program on control and safety rods followed two lines of approach; first the test of scaled-up units of the type proven in SRE experience and involving separate control and safety rods; and second, the development of

units which combine the functions of shim control and safety. This latter approach is highly desirable, since it reduces the total number of units in the core; hence it was selected for the Hallam plant. In order to obtain the necessary design information required for this concept, an in-pile test was run in the SRE employing a special test assembly. Cylindrical poison rings of several materials and of various dimensions and clearances within the thimble were irradiated with temperature and flux instrumentation to confirm analytical calculations of temperature profiles within the poison material. The alternate poison materials reviewed included boron-nickel alloy, boron carbide, hafnium, hafnium oxide, and several rare-earth oxides.

MECHANICAL PUMPS

Full-sized prototype centrifugal pumps of two types, each having a capacity of 7200 gpm at 150 feet sodium head, were procured for testing. One type employed a frozen sodium shaft seal and the other type involved a free sodium surface and incorporated a hydrostatic radial bearing supplied with sodium through a bypass line from the pump discharge. The free surface pump was finally selected. The prototype pumps were installed in separate loops of 12-inch pipe size which contained the valves, flowmeters, and instrumentation required to determine pump characteristics and performance.

SYSTEM PREHEATING

A development program was conducted to determine the most economical method of preheating the large (14- and 16-inch) Type 304 stainless-steel pipes. Transformer heating and 60-cycle induction heating experiments were conducted. The transformer heating experiment consisted of a balanced-bridge circuit which was designed to eliminate stray currents at the grounded points of the piping loop. This method of heating was successful for the empty pipe, but did not economically melt solidified sodium in large pipes.

Because Type 304 stainless steel is austenitic, it was necessary to clad the pipe with carbon steel in order to effectively use 60-cycle induction heating. Several thicknesses of carbon steel were wrapped around the stainless steel

piping to determine optimum cladding thickness. The temperature distribution along the pipe was not uniform for any thickness, and this system was not suitable for 3-phase operation. As a result of these experiments, it was decided to use tubular resistance type heaters for the entire Hallam plant. In order to assure uniform temperatures, the heaters will be placed in a 1-inch oven between the heated component and the insulation. This furnace type system will be used for pipes, valves, pumps, tanks, and all other sodium system components.

SODIUM INSTRUMENTATION

Specialized instrumentation is required to measure level, pressure, and flow of the sodium at various points in the reactor plant. The development program extended existing instruments to the larger sizes required and attempted to obtain higher reliability and lower temperature sensitivity than is presently obtainable with the instruments used in the SRE. Progress was made in the development of high-temperature insulation, permitting level coils to operate for extended periods at temperatures of 1000°F. Considerable effort was required on flow meters for the indicated 14-in. piping size in order to develop accuracy over the full flow range and to eliminate calibration changes with time due to temperature. Work on an improved air-cooled plugging meter device used for determining the sodium oxide content was also completed.

VALVES

In the SRE, valves up to six inches, employing both bellows seals and frozen sodium stem seals, have been used. Operating experience with valves, which were largely of conventional design with some minor modifications, indicated the desirability of incorporating improved features for the larger valves up to 14 in. required for the HNPF. By procurement and test of a number of standard commercial and specially designed valves, the best design features of each were determined and incorporated in the specifications for the production valves. The valve test program involved determination of across-seat leakage, both before and after thermal cycling, stem leakage, and life tests of the valve operators.

STEAM GENERATOR

MSA Research Corporation was contracted to test a model of the Hallam steam generators. The model consisted of a kettle-type evaporator, an eliminator, a nineteen-tube superheater, and a two-tube superheater.

The units were constructed of a low-alloy steel material with bayonet-duplex tubes for liquid metal service and were installed and tested in a 3000-kw (thermal) liquid metal test loop.

The operational test program, which started on April 14, 1959, consisted of 156 steady-state heat transfer runs, vapor purity tests, 100 10-100-10% power cycles, 500 10-80-10% cycles and 26 dynamic performance studies. The test program was completed on October 16, 1959 after 3615 hours of operation. Since the tube bundles used in this test program had previously been operated (under another program) for 4926 hours, their accumulated test time is 8541 hours. The successful test of this model has demonstrated the integrity and performance of this type of steam generator under severe operating and test conditions.

COLD TRAPS

Operation of the SRE has indicated several areas where significant improvements can be made in the cold traps (which are used to reduce the oxygen content of the sodium coolant to an extremely low value). In the primary sodium system, as a result of the free sodium surface in the core tank, some contamination with oxygen is unavoidable during handling operations on core components. For this reason, the Consumers prototype cold traps are designed to operate efficiently and rapidly to reduce the oxygen content of sodium. In order to determine the operating performance of the cold trap, a full-scale prototype was tested in a loop in which oxygen was continuously added to the sodium before it entered the trap, in this way simulating the conditions that would prevail during initial cleanup of a large quantity of sodium.

Figure 3. Reactor Building

REACTOR BUILDING

The reactor building houses the reactor, heat transfer system, reactor service systems and equipment, and miscellaneous maintenance and service areas. The building design conforms to all applicable building codes. The building is of steel frame construction with insulated roof deck and light-gauge, fluted, insulated, and protected metal siding. The roof and walls are designed to form as tight an enclosure as is possible using conventional materials.

The building is 275 ft long by 116 ft wide. The 116-ft width of the building comprises two bays, an 81-ft wide by 70-ft high bay over the reactor and an auxiliary bay 35 ft wide by 51 ft high. Overhead cranes of 60 tons and 25 tons respectively service each of the bays. The operating floor of the reactor plant is at finished plant grade elevation of 1440 ft.

Beneath the operating floor, in shielded compartments, are the sodium graphite reactor, the radioactive portion of the sodium heat transfer system, fuel cleaning and storage facilities, the maintenance cell, and piping and service equipment. These shielded compartments are of regular concrete having adequate thickness to fulfill the shielding requirements. High density concrete is used above the reactor and in the operating face of the maintenance cell.

These compartments, housing radioactive sodium equipment, are lined with steel to prevent leakage of the nitrogen atmosphere. The concrete access plugs to these areas are gasketed to prevent nitrogen leakage, and all plugs are strapped to preclude radiation streaming. A railroad spur track enters the high bay area at the south end of the building to facilitate the delivery of heavy components and for shipment of spent fuel elements to the reprocessing plant. The tank car sodium melt station is adjacent to the railroad spur, at the south end of the low bay area.

The steam generators and associated components of the nonradioactive portion of the sodium heat transfer system are housed in three separate enclosures at the north end of the high bay area. The high bay bridge crane has coverage over the ceilings of the steam generator rooms.

A lead shielded fuel handling cask weighing approximately 200 tons is mounted on a gantry which operates in the reactor room between the reactor and the fuel cleaning and storage area. The control rod drive mechanisms are mounted on an integral movable support structure. The drive mechanisms can be disconnected from the control rods at the upper surface of the reactor loading face shield and the entire assembly can be moved away, leaving the shield face clear for fuel handling operation.

Adjoining the reactor plant at the north end is a central control room serving both the conventional plant and the nuclear facility. The offices, the equipment decontamination room, and the plant ventilation equipment room adjoin the reactor plant to the west.

The radioactive waste facility is located in a separate building to the west. The objective of the radioactive waste system is to collect, store, and monitor any potential radioactive liquid or gaseous wastes for either decay and ground disposal or future processing and packaging, as required.

The subsoil at Hallam consists of homogeneous layers of stiff clay extending to a depth of some 80 ft below finish grade. The soil is remarkably uniform, having unconfined compressive strengths ranging from 8,000 to 15,000 pounds per square foot. A criterion of 6,000 pounds per square foot was used in design to allow for uneven settlement of the finished structure.

The steel superstructure is designed to the American Institute of Steel Construction requirements with wind loads up to 40 pounds per square foot and seismic conditions of 10% gravity loads.

BUILDING VENTILATION

The reactor building ventilation system is designed to satisfy general heating and ventilation requirements for personnel comfort and to minimize the possibility of contaminating the entire plant as a result of an incident in which radioactivity is released to the air. The design objective with respect to confinement of the building atmosphere is to maintain a negative pressure (relative to outdoor barometric pressure) to ensure that leakage of air is inward toward potentially radioactive areas, at 320% of building volume in 24 hours at a negative pressure of 1/8 in. of water.

Figure 4. Reactor Building Layout
Plan above Elevation of 1460 ft

Figure 5. Reactor Building Layout
Plan below Elevation of 1460 ft

Figure 6. Reactor Building Layout
Plan below Elevation of 1440 ft

Figure 7. Vertical Sections through
Reactor Building

Figure 8. Current Stage of Completion within the Reactor Building

Most areas of the building are continuously supplied with filtered and tempered air on a once-through basis from a central comfort heating system. The air is then filtered for particulate matter, and released to the atmosphere through the building vent stack. In the control room, viewing gallery, counting room, and amplifier areas, the room air is maintained at a constant pressure slightly above outdoor barometric pressure, at a constant temperature of 70°F, and at a maximum relative humidity of 50% by means of unit type air conditioning systems. In the steam generator rooms, the battery room, and in certain service areas, the air is supplied from a central comfort heating system but is exhausted directly to atmosphere.

The ventilating rate requirements throughout the building vary from two volume changes of outside air per hour in the reactor room to 20 changes per hour in such areas as the laundry and the battery room.

The possibility of spreading airborne contamination is minimized by maintaining a pressure differential within the building so that the flow of air is toward areas more likely to become contaminated. Clean areas, such as the control room, in which no contamination is anticipated, are maintained at a water pressure of +1.8 in. with respect to atmospheric pressure. Areas in which no contamination is expected, but where the possibility exists, are maintained at pressures varying from atmospheric pressure to 1/4 in. of water below atmospheric pressure, depending on the probability of contamination. The maintenance cell is normally contaminated, and is kept at 1/2 to 3/8 in. of water below atmospheric pressure.

An integrated differential pressure control system automatically regulates the static pressure and the volume change rate of ventilation air in each area. Motorized dampers control the flow of air in the exhaust branches. Supply is manually controlled. The dampers are operated by proportioning controllers equipped with pressure piping open to the room atmosphere and reference piping open to the outside atmosphere.

Building exhaust through the stack is continuous at all times. This is accomplished by having 100% spare exhaust fan equipment. In the event of a power failure, the normal and spare fans are supplied from the diesel generator unit. All effluent is filtered for removal of particulate matter, and the stack is provided with a radiation detection and alarm system.

FIRE PROTECTION SYSTEMS

The fire protection systems throughout the plant are designed to combat the specific types of fires most likely to be encountered in any given area.

Fire protection outside of the building is provided by water hydrants and hose cabinets. Throughout the building, various types and sizes of portable carbon dioxide and chemical applicators are provided. In the vital electrical equipment areas, such as control room, amplifier equipment rooms, cable spreading room, steam generator room, and diesel engine room, portable carbon dioxide extinguishers are provided. In non-sodium areas, such as ventilating equipment and decontamination equipment rooms, automatic sprinkler systems are provided. In the sodium areas beneath the reactor room floor, and in the maintenance cell, the nitrogen and helium atmospheres which provide operational inert gas blankets, also provide fire protection. Supplementary protection for these sodium areas is provided by hand and wheel-type dry chemical systems. In addition, these and all other sodium areas are provided with dry bulk calcium carbonate which can be applied to small sodium spills with hand shovels.

Fire detection and alarm instrumentation is installed throughout the plant to minimize the fire hazard.

REACTOR STRUCTURE

The primary purposes of the reactor structure are to provide a vessel to contain the reactor core and the associated liquid sodium coolant, to provide a firm and level support for the reactor components, and to provide adequate shielding against radiation from the reactor.

The guiding philosophy in the design of the components of the reactor structure is that of obtaining safety with conservation. This is necessary not only from the standpoint of minimizing hazardous conditions, but also because of the extreme measures required for repair after operation begins. It is of the utmost importance that the integrity and usefulness of the reactor structure be maintained through all anticipated and unanticipated operating conditions. The

most economical design was strived for; however, the economic considerations were frequently tempered by design conservation.

Components of the reactor structure in contact with liquid sodium are made of Type 304 stainless steel for corrosion resistance, strength, and low creep rate at high temperatures. The exceptions to this are the loading face shield and core clamp assembly, where a ferritic stainless steel is used because of its low coefficient of expansion. Type 405 stainless steel was chosen because of its welding characteristics, which are better than those of most other ferritic stainless steels. Those components not normally in contact with sodium are made of carbon or low alloy steel.

Conservatism in the structural analysis was achieved by maintaining relatively low stress levels within the reactor structure components, as well as by minimizing the degree and number of stress raisers

The reactor structure is composed of the following general components:

- a) Loading face shield
- b) Reactor vessel and internals
- c) Containment vessel
- d) Thermal shields and thermal insulation
- e) Support structure for vessels, thermal shields, and insulation
- f) Cavity liner and reactor cavity diaphragm seal

These components are located in the reactor cavity below the floor of the reactor room and are described below:

Loading Face Shield. The loading face shield is a large, circular, stepped plug of reinforced dense concrete encased in ferritic stainless steel. The diameter at the bottom of the plug is 18 ft 6 in. and increases 8 in. at the step to 19 ft 2 in. Overall depth is 7 ft 3 in. A support flange at the top extends to a diameter of 21 ft 2 in. and rests on 40 rollers equally spaced around the periphery of the shield to permit rotation. The total weight of the shield is approximately 300 tons. To provide reactor atmosphere containment, the loading face shield is sealed to the upper cavity liner by a frozen metal alloy seal. Cooling pipes embedded in a layer of lead between the concrete and the bottom plate keep the bottom plate and concrete temperatures within acceptable limits.

Figure 9. Cutaway View of the Reactor Core

Figure 10. Reactor Vessel Nearing Completion
(Published through courtesy of Baldwin-Lima-Hamilton Corp.)

The shield is penetrated by vertical stepped sleeves, through which the core elements are inserted into the reactor. The sleeves are sealed by matching shield plugs.

Reactor Vessel and Internals. The reactor vessel, which is attached to the upper cavity liner by the bellows, contains the reactor core, its associated pool of liquid sodium, and the sodium vapor and helium gas atmosphere above the liquid pool.

The reactor vessel is essentially an open-top flat-bottom vessel 19 ft in outside diameter by 33 ft in inside height. The lower 42 inches of the reactor vessel below the grid plate forms the inlet plenum. Here, three 14-inch sodium lines return the relatively cool sodium (607°F) from the intermediate heat exchangers to the reactor. This is the highest-pressure plenum of the vessel and has a thickness of 2 inches. A short distance above the grid plate support ring the vessel wall is reduced from 2 in. to 3/4 in., the outlet nozzle course is again increased to 2 in. to resist the piping reactions from the three 16-in sodium outlet lines which carry the hot sodium (945°F) back to the intermediate heat exchanger. The wall thickness above the outlet nozzle course to the bellows is 1 in. A 1-in.-thick grid plate separates the inlet plenum from the rest of the vessel above. This grid plate rests on a support ledge in the reactor vessel. Away from this support ledge, the grid plate is supported by 141 moderator element support rods. Attached to the grid plate above each rod is a moderator support pedestal which positions the moderator and reflector elements in relation to the loading face shield overhead.

Nozzles for the fuel process tubes extend through the grid plate and provide the flow from the inlet plenum to the reactor core.

The total weight of the reactor vessel with its internal structure is approximately 180 tons.

The reactor vessel alignment will be maintained by keys on the lower periphery of the vessel which fit into keyways in the containment vessel.

Containment Vessel. The containment vessel limits the drop in sodium level in the reactor core in the event of a major sodium leak interior to the

Figure 11. Grid Plate with 141 Moderator Element Supports Nearing Completion
(Published through courtesy of Baldwin-Lima-Hamilton Corp.)

primary blocking valves. The containment vessel will maintain a sodium level sufficient to assure natural circulation in the core.

The containment vessel is 20 ft 8 in. in inside diameter by 35 ft 3 in. in inside height. Guard pipes surround the primary pipes and extend from the containment vessel to the primary blocking valves. The side walls of the vessel are 1/2 in. thick, and the bottom head is approximately 1-1/2 in. thick. The vessel and guard pipe is fabricated from low alloy carbon steels. Ten equally-spaced, full-length T-section brackets are welded to the inside wall of the containment vessel to provide horizontal and lateral support of the thermal shields.

The containment vessel alignment will be maintained by keys on the lower periphery of the vessel which fit into keyways on the cavity liner.

Directly beneath the bottom of the containment vessel, permanently-mounted heater elements are provided to maintain reactor temperature during prolonged shutdowns and to aid in the preheating of the reactor structure.

The total weight of the containment vessel is approximately 50 tons.

Thermal Shields and Thermal Insulations. Thermal shielding is required to attenuate gamma and neutron radiation in order to prevent the concrete biological shielding from overheating. The thermal shielding is provided by two 2-3/4 in. -thick concentric cylinders of carbon steel located in the annular space between the reactor vessel and the containment vessel. It is built up from segments which are held in position by the "T"-shaped brackets on the containment vessel. These brackets keep the thermal shields in proper alignment while permitting radial and vertical expansion.

The total weight of the thermal shield is approximately 225 tons.

Block thermal insulation is mounted on the outside periphery of the containment vessel. This consists of 12 in. of Superex insulation.

The top of the cavity is insulated with similar Superex insulation wherever it could be exposed to heat radiating from the thermal shields or the reactor vessel. Also, there is block insulation between the containment vessel and the cavity liner, in and around the support columns.

The total weight of insulation is approximately 19 tons.

Support Structure for Vessels, Thermal Shields, and Insulation. A firm and level support structure is provided for the containment vessel, which in turn supports the reactor vessel, thermal shields, and insulation. The support structure is in the form of many short pipes, standing on end as columns with a combination shim and cap on the top. The cap will be machined to provide any shimming required for leveling. The pedestals are welded to the cavity liner. The pipes are 18 inches in outside diameter. The columns plus the shim caps give a nominal height of 25 inches. There are 79 support columns provided for the reactor.

Cavity Liner and Reactor Cavity Diaphragm Seal. The cavity liner configuration results from the design of the concrete biological shielding and the configuration of the piping. The inside dimensions of the cavity liner are established by the minimum clearance required for installation of the reactor structure.

The previously-described reactor structure components are located within the reactor cavity, which is sealed at the top by the loading face shield. A 1/2-in.-thick (one inch on bottom and top above and below reactor) carbon steel liner is provided which extends into the pipe tunnels to the diaphragm seals. The cavity itself is about 40 feet deep with an inside diameter of 24 feet. The four pipe tunnels extending out from the reactor cavity are 12 feet high by 5 feet wide. The cavity liner provides a gas-tight envelope between the reactor cavity and the outside concrete structure. This envelope is needed to prevent moisture from entering the reactor cavity as well as to enclose the dry helium atmosphere surrounding the reactor structure. The liner will also contain sodium vapors in case of a leak from the reactor vessel or primary piping.

Cooling is provided by water circulating through 1-in. pipes on the exterior surface of the liner to remove heat generated in the concrete and cavity liner by the attenuation of the radiation from the reactor, as well as sensible heat.

The reactor cavity diaphragm seals located just interior to the primary blocking valves form the seal separating the reactor cavity with its helium atmosphere from the pipe galleries with their nitrogen atmosphere.

Figure 12. Reactor Cavity Liner

SUMMARY OF WEIGHTS AND LOADS

	(lbs)	
Reactor and internals	156,200	
Thermal shield	450,000	
Containment vessel	101,000	
Insulation	38,000	
Sodium	252,500	
Moderator cans	<u>352,500</u>	
Total	1,350,200	= 675 tons load carried by the support structure
Top shield	600,000	
Fuel, control rod, and miscellaneous elements supported by top shield	<u>255,000</u>	
	855,000	= 427 tons total

REACTOR CORE

The primary objective calls for design of a thermal-heterogeneous reactor core which utilizes individually-contained graphite moderator elements, slightly-enriched U 10 - Mo fuel, and liquid sodium coolant. Control will be obtained by the insertion or removal of poison columns into or out of the core. Sufficient nuclear power is to be produced in the fuel and transferred to the sodium cooling system at a temperature level such that, with the auxiliary heat transfer system, 725,000 lbs per hour of steam at 800 psia and 825°F will be available at the turbine throttle. This requires a thermal reactor power level of 254,000 kw of heat and a bulk sodium outlet temperature of 945°F.

A secondary objective calls for provision of sufficient flexibility in the design of the core that alternate fuels, other than U 10 - Mo, can be used. Such an alternate fuel might be uranium metal alloys, thorium or thorium oxide fuel, or uranium carbide.

The core design is basically similar to that of the SRE. Fuel is suspended from the top shield in a matrix of moderator elements, with fuel and other core elements being individually loaded through the top shield by the fuel handling cask. The hexagonal moderator elements, which are completely submerged in sodium, are scalloped at each corner. When stacked together, three scalloped corners form a circle in which process channels for fuel and control rod thimbles are located. All moderator elements are supported and located upon grid plate pedestals, which transmit the element weight to the core support structure.

The major components of the reactor core are the fuel elements, moderator and reflector elements, reflector filler and dummy elements, core clamps, safety rods, neutron source, sodium level measurement elements, and moderator temperature elements.

FUEL ELEMENTS

The fuel elements consist of an 18-rod fuel element assembled integrally within a process tube. The assembly is designed to be supported by means of a hanger tube from a plug which is within the top shield.

FUEL ROD

Figure 13. Fuel Element Assembly

Figure 14. Moderator-Reflector Element

The total active length of the fuel is 13 ft, 6 in. The fuel is in the form of 0.590-in. diameter slugs which have been inserted in 0.010-in. thick, 0.660-in. OD, Type 304 stainless-steel tubing. Each rod is provided with 21 in. of expansion space at one end only.

532 039

MODERATOR AND REFLECTOR ELEMENTS

The moderator cell is a canned graphite log, hexagonal in cross section, with corner scallops.

REFLECTOR FILLER AND DUMMY ELEMENTS

The graphite density in the reflector region is increased by filling the scalloped corners of the moderator log with a circular canned graphite log.

932 040

REACTOR DESIGN DATA

The design data for the Hallam Reactor are listed below:

Power	256 Mwt
Fuel	U - 10 wt % Mo
Moderator	Graphite
Primary coolant	Sodium
flow rate	8.4×10^6 lb/hr
inlet temperature	610°F
outlet temperature	945°F
Secondary coolant	Sodium
flow rate	8.4×10^6 lb/hr
IHX inlet temperature	565°F
IHX outlet temperature	900°F
Active core height	13.5 ft
Active core diameter	12.9 ft
Moderator and reflector height	17.0 ft
Moderator and reflector diameter	16.8 ft
<u>Fuel</u>	
Fuel elements	137
Uranium inventory	27,600 kg
U ²³⁵ enrichment	3.6 wt-%
U ²³⁵ inventory	994 kg
Maximum fuel burnup	15,000 Mwd/MTU
Radial peak to average power	1.4
Axial peak to average power	1.65 max. 1.5 avg.
Average specific power	247 kg
Average fission density in fuel	4.4×10^{12} fiss./cm ³ sec
Process tube material	Zircaloy 2

Fuel Elements

Fuel rods per element	18
Fuel slug diameter	0.590 in.
Sodium bond annulus	0.025 in.
Stainless steel cladding thickness	0.010 in.
Fuel rod OD	0.660 in.
Average fuel rod power to power of hottest rod	0.88 nominal (0.86 minimum)
Flow adjustment permitted by variable orifice	4:1

Central Fuel Element Characteristics

Design power	2.46 Mwt
Design maximum fuel temperature	1250°F
Nominal maximum fuel temperature	1184°F
Maximum surface heat flux	3.8×10^5 Btu/hr ft ²
Maximum rod power	19.4 kw/ft
Coolant flow area	7.35 in. ²
Core pressure drop	9.6 psi
Sodium average velocity	8.6 ft/sec
Sodium flow rate	22.5 lb/sec

Moderator and Reflector Elements

Moderator	Hexagonal graphite logs, edges scalloped to provide process channels
Moderator cladding	Type 304 stainless steel
Size across flats	16 in.
Length	17 ft
Graphite density	1.68 gm/cm ³
Stainless steel thickness	0.016 in.
Sodium gap between elements	0.160 in. avg.

Control Rods

Number of driven shim-safety-regulating rods	19
Number of stationary rods (temporary)	6
Thimble material	Zircaloy-2
Control rod atmosphere	Helium, 15 psig
Control rod travel	13.0 ft
Poison column OD	3.4 in.
Poison column length	12.5 ft
Poison material	45% Gd_2O_3 - 45% Sm_2O_3
Poison cladding material	Hastelloy X
Maximum poison column temperature	1800 °F
Reactivity worth of driven rods	14 12% $\Delta k/k$
Maximum reactivity rate	0.03%/sec

Steam Conditions

	Design Obj.	^{Base} 800	High Power
Throttle steam pressure, psig	800	800	850
Throttle steam temperature, °F	825	843	865
Throttle steam flow, lb/hr	710,000	753,000	839,000
Net electrical output, Mwe	75	~80	~87
Gross electrical output, Mwe	81.8	80	95.0

* Corresponds to 256 min.

932 043

Figure 15. Sodium Heat Transfer Circuit

932-044

SODIUM SYSTEMS

The sodium systems consist of the sodium heat transfer system and the sodium service systems.

The sodium heat transfer system has the following basic objectives:

- a) To transfer thermal energy from a reactor heat source to a non-radioactive steam system which is located in an accessible area external to any radiation field.
- b) To provide coolant to the core for removal of afterglow heat during a reactor shutdown.
- c) To respond to the demands of a plant control system and a plant protective system.

The sodium heat transfer system consists of three parallel circuits originating at the reactor and terminating at the steam generators. Each sodium circuit consists of a radioactive primary loop and a nonradioactive secondary sodium loop, both of which exchange thermal energy through an intermediate heat exchanger.

The three primary sodium loops are parallel loops common to the reactor, and the intermediate heat exchangers for each loop are located in individual shielded cells. In each primary sodium loop there is a variable-speed primary sodium pump located in the same cell as the intermediate heat exchanger. The drive for the pump is located outside of the shielded cell above the reactor floor level.

The secondary sodium system consists of three separate closed loops originating on the shell side of an intermediate heat exchanger and terminating at the sodium side of a steam generator. Each loop contains a variable-speed secondary sodium pump, expansion tank, throttling valve, pressure and temperature transmitters, radiation detector, and electromagnetic flow meter and interconnecting piping. In addition, several small isolation valves between the sodium service system and each of the separate secondary sodium loops are included in the secondary sodium system. Only that portion of the piping in a loop which connects to an intermediate heat exchanger extends into the shielded cells. The balance of each loop is in an accessible, unshielded area above floor level.

The sodium heat transfer system is designed as a load following system. That is, the reactor power level and sodium flow rates are varied according to turbine load. Safety instrumentation is provided throughout the reactor system to reduce power level and sodium flow or, if necessary, shut down the plant should a potentially dangerous condition arise.

The sodium service systems provide the means to fill, drain, flush, and purify the sodium in the reactor core tank and the sodium heat transfer systems.

INSTRUMENTATION AND CONTROL

The reactor instrumentation can be divided into nuclear instrumentation, sodium instrumentation, plant control system, plant protective system, and radiation monitoring system. The nuclear instrumentation circuits measure and indicate (or record) neutron flux over the entire operating range of the reactor, starting from source level. The sodium instrumentation measures and indicates (or records) pressures, flows, temperatures, and liquid levels in the sodium systems. The plant control system operates the reactor plant (adjusts control rods and pump speeds) in accordance with plant load demand either by manual (semiautomatic) or automatic actuation. The protective system, consisting of alarm, setback, and scram circuits, safeguards the reactor against malfunctions and errors which might otherwise create hazardous conditions. The radiation monitoring system measures radiation levels throughout the plant.

The objective of the nuclear instrumentation system is to continuously monitor core neutron density and generate signals representing core power, neutron level, and reactor inverse period. These signals are used in plant operation, control system, and protective system functions during core loading, startup, shutdown, and power operations.

The nuclear instrumentation is subdivided into nine channels of permanently-installed instrumentation and three channels of mobile in-core instrumentation. In the nuclear instrumentation system, the neutron detecting instruments are located in tubes which extend down from the reactor operating floor to a position at the cavity liner near the bottom of the reactor core.

Nuclear instrumentation system operation can be conveniently grouped into core loading, plant startup, and power operations. Each type of plant operation is unique with respect to the operational problems presented.

In reactor startup, the log count rate is recorded and the period measurements are indicated. The period signals tie into the plant protective system and will initiate an alarm and a subsequent setback if the period becomes shorter than preset limits. The nuclear instrumentation also transmits signals to the plant protective system which are used in the power flow and reactor scram,

setback, and alarm circuits. Other signals from this system are directed through amplifiers to the neutron flux controller which positions the shim regulating rods in response to plant load demands.

SODIUM INSTRUMENTATION SYSTEM

The objective of the sodium instrumentation system is to detect, display, and record (or indicate) all necessary temperatures, flows, levels, and pressures in the sodium heat transfer and service systems.

Sodium instrumentation systems are designed to meet several basic criteria:

- a) To withstand temperatures up to 1200°F
- b) To withstand high radiation levels
- c) Sodium instruments are designed to provide a minimum of contact between the moving parts and the sodium and to allow no thin or weak material sections. When either of these are necessary, double containment is provided.

The sodium instrumentation measures temperatures, liquid levels, flow rates, and pressures. The sodium exit temperatures from the reactor core are measured and transmitted to the plant protective system to initiate setback or scram in the event of excessive fuel-channel sodium outlet temperatures. The sodium liquid level in the reactor vessel is measured by means of induction coils over the range expected during normal operation. High- and low-liquid-level alarms and scram circuitry are provided. Sodium flow rates are measured by magnetic flow meters. The pressure-sensing elements are NaK-sealed, utilizing diaphragms and bellows.

The design of the plant control system is based on automatic operation and load following between 15 and 100 percent of design electrical power, semi-automatic capability at all power levels, and maintenance of temperatures and pressures within their design limits at all times.

This plant is one of a new generation of nuclear facilities, and the first sodium-graphite reactor having the sole purpose of power production. As a

consequence, safety is of prime concern, but the economic production of power is also an important consideration. To this end, the plant protection system has been separated from the plant control system where physically possible (the exceptions are few).

The control system is based on the largest possible use of proven and commercially available components in order to minimize down-time due to instrument failures. Standard pneumatic controllers are extensively used. However, to minimize shut-down by the protective system, careful attention has been paid to error monitoring and displays, rate limiting, automatic-manual transfer operations, and similar items.

The plant is designed as a load-following system - that is, reactor power level and sodium flow rates are varied according to steam demand. An increase in plant load opens the turbine throttle valves, which results in increased steam flow and decreased pressure. Signals representing both flow and pressure are transmitted to the sodium pumps, increasing their speed proportionately. The increase in sodium flow through the reactor acts on the neutron flux controller to withdraw the shim regulating rods, increasing the reactor power level.

A conventional, three-element, feedwater control system proportionately increases the feedwater to the steam generators. In the event of a sudden load decrease too fast for the pressure control system to follow (such as a turbine trip) a back-pressure controller dumps the steam direct to an atmospheric boiler.

REACTOR PLANT PROTECTIVE SYSTEM

The plant protective system protects the reactor and process systems from thermal damage during all phases of plant operation. The various incidents which can lead to thermal damage, if left uncorrected, are:

- a) Failure of the plant control system.
- b) Operator errors while in manual or semi-automatic control (e. g., inadvertent control rod withdrawal or change in pump speed).
- c) Component failure or malfunction not included in the control system (e. g., loss of eddy - current coupling or failure of a feedwater valve).
- d) Partial or total loss of electrical power to the plant.

The entire plant protective system consists of several essentially independent subsystems which are themselves composed of the electrical, electronic, and electromechanical components necessary to fulfill the system objectives. The four main subsystems are scram, setback, alarm-annunciator, and interlock.

The operation of the HNPF plant protective system is probably best described by an indication of where the system offers adequate protection while the plant is being operated in the variety of conditions under which it is expected to perform. Principal among these are:

Plant Startup

Startup operation is defined as the process of bringing the reactor into the power range from a condition in which all the control rods are initially inserted. Startup can occur from a wide variety of operating conditions. The main concern during startup operation is the possibility of attaining an extremely short reactor period before corrective action can be taken.

Plant Warmup

After the reactor has been successfully brought to a critical condition and on a constant safe period, the power level is brought up into the power range and leveled off at a reasonable low value ($<100\text{kw}$). From this point on, power level is increased slowly (either on manual or semiautomatic control) until reasonable system temperature increase rates are attained.

The plant protective system takes corrective action automatically. This action can be in the form of reducing power or scrambling the reactor. When a scram occurs, the immediate and most important action is the dropping of the safety rods into the core to make the reactor subcritical. Simultaneously, other automatic actions are initiated to shut down the plant in an orderly and safe manner. This system is designed with fail-safe circuits and equipment wherever possible.

RADIATION DETECTION AND MONITORING

The radiation detection and monitoring system provides protection for personnel; i. e., indications of off-normal conditions and prevention of the release of excessively radioactive material to the plant environs. This installation provides gamma monitoring in areas routinely occupied by operating personnel and

gamma monitoring in areas of higher radiation level that are infrequently occupied. Should radiation levels become excessive, alarms are given and automatic action initiated where required.

Figure 16. Fuel Handling Machine

FUEL AND COMPONENT HANDLING

This system provides for receipt, handling, cleaning, storage, and shipment of fuel elements, control rods, moderator and reflector elements, and other replaceable core components required during startup or normal operation of the reactor. Facilities are also provided for handling, decontamination, or shipping, as required, of such replaceable noncore components as pumps and cold traps.

Two general cycles are involved in the handling of fuel and core components:

- a) Refueling of reactor
- b) Moderator element replacement

A refueling system replaces fuel elements in the reactor at periods to be finally established by a fuel management program. Current objectives are to have the fuel change period coincide with the periodic plant maintenance and inspection schedule. The latter period, approximately two weeks, will exceed the time required to effect the refueling. The number of fuel elements to be replaced will be established by maximum permissible burnup of the fuel and the load factor.

The two-week shutdown period will allow time for clearing of the top shield, reducing the sodium temperatures, decay of fission products to limit fuel element temperatures after withdrawal from sodium, replacement of fuel elements, and re-establishing criticality and sodium temperatures.

The fuel handling system removes fuel and other core components without releasing radioactive contaminants into the reactor building.

The storage system provides sealed and shielded storage space for all replaceable reactor elements. Cooling is provided for spent fuel elements. All of vault No. 2 and vault No. 3 are required for storage if all small core components are removed from the reactor for moderator element change. Fuel element cladding ruptures are detected by sampling the fuel storage cells for fission gases by means of a portable purge unit.

The cleaning system removes sodium and sodium compounds from pumps and core components prior to maintenance and operation preparatory to shipment off-site.

FUEL AND MODERATOR HANDLING MACHINE

The fuel and moderator handling machine consists of a shielded cylinder with a gas lock, indexing device, and a hoist and grapple mechanism to raise and lower components. The machine is supported and transported by a trolley which in turn rides on a gantry crane.

WASH CELLS

There are four steel-lined cleaning cavities approximately 40 ft deep. Three of the cavities are equipped with removable cells for washing fuel elements, small core components, and moderator elements. The fourth cavity is equipped with a removable cell for washing pumps. A fourth fuel wash cell with larger diameter adapter rings and guide vanes can be installed in the pump wash cavity to provide another fuel wash facility, if necessary. Fuel wash cells have cooling water jackets to remove decay heat from irradiated fuel elements. The cooling jackets have a capacity of 73,000 Btu/hr, which is based on the heat generated in a center element 10 hours after shutdown.

STORAGE CELLS

Shielded storage cells are provided for storage of core components. The small core component storage cells are suspended in three vaults below the operating floor. The first vault is for storing nonradioactive miscellaneous core components such as core heaters and spare core components. The second vault is used when the complete core is removed from the reactor, and the third vault is used for normal fuel change. The cells in vaults No. 2 and 3 will store spent fuel elements, control rods (with control rod thimbles or by using the control-rod storage-plug sleeve adapter), dummy elements, neutron detection and flux instruments, source capsule assembly, source thimble assembly (Be), moderator element measuring instrument, sodium temperature measuring elements, and liquid level measuring elements. The cells in vault No. 3 will normally be utilized to store new fuel elements prior to their placement in the reactor. The cell length is long enough to accommodate the longest reactor element, and the diameter is large enough for fuel elements and control rods.

PICK-UP CELL

The pick-up cell will be utilized for transferring to the fuel handling machine a new fuel bundle in a shipping container suspended from a shuttle rod in the cell. The cell will not be used to handle radioactive fuel, and thus cannot become contaminated. Nitrogen and vacuum services are required to evacuate and fill the cell with nitrogen before transferring the fuel bundles to the machine.

MAINTENANCE CELL

The maintenance cell is designed for the disassembly and reassembly of radioactive components. It consists of two defined areas which may be designated as the in-cell area and the operating area.

The in-cell area is 7 ft wide, 9 ft long, and 54 ft deep from the reactor room floor level. An offset area 18 ft deep from the reactor room floor level extends the basic cell to 9 ft wide by 12 ft long. A wall storage area is located at the end opposite the viewing windows, and it is 2 ft by 7 ft by cell depth. A recessed area located at the top of the cell is provided for installation of the in-cell crane.

The operating area is 6 ft wide, 16 ft long, and 51 ft deep from the reactor room floor level. The area is divided into three floor level. The area is divided into three floor levels located at 16 foot intervals and 5-1/2 ft below the center line of each of the three viewing windows. A stairway leads from the reactor room level to each of the operating area levels. Each floor level and the roof of the operating area has a removable floor section for removing equipment from the area.

A control console is located on each of the levels of the operating area for control of the in-cell equipment. The consoles are located in front of the viewing windows. A master control board is provided at the first floor station that permits operation of only one station at a time.

SHIPPING CASKS

Two shielded casks are required for shipping and handling of radioactive components:

1. The fuel and general purpose shipping cask is designed to contain either fuel elements, moderator element, control rods, poison rings, or radioactive solid waste. It can be modified to accommodate any item that can be contained in the moderator or fuel element shipping canister.
2. The pump and cold trap handling cask is used for handling pumps and cold traps. When a pump or cold trap is shipped, the appropriate handling cask will be used but will be modified by adding an internal shielding sleeve and bottom head. It will also be necessary to support the cask, during shipping, with a cradle.

The fuel and general purpose cask consists of a series of spools, a top, a bottom, and a liner. The spools are made of concentric steel cylinders with the annular spaces filled with lead. The ends of the spools are stepped for shielding purposes. The stepped ends fit together, and a seal between the joined spools is made by means of flanges and bolts similar to those used in piping practice.

Fuel elements are shipped to another facility for recovery of uranium and plutonium. Before shipment, fuel elements are stored for one hundred days to permit sufficient decay of fission products to facilitate chemical processing and to reduce shielding requirements. The total cooling time of each fuel element prior to reprocessing will be from 120 to 130 days.

PORTABLE PURGE UNIT

The portable purge unit consists of a cylindrical glove box with a hydraulic jack and grapple, hand-operated hydraulic pump, gloves, glove port cover plates, 30-in.-Hg - 15-psig compound pressure gage, flexible hoses, valves, and couplings. The hydraulic pump is carried on a cart which is also used to transport the glove box and hoses. The upper part of the glove box is made of a cast acrylic section that permits vision within the box. Helium and vacuum connections are provided on the glove box which may be connected by means of the flexible hoses to the helium and vacuum connections in the floor.

932 056

AUXILIARY SYSTEMS

Wherever molten sodium contacts a gaseous atmosphere such as is found in the top of the reactor, sodium storage tanks, free surface pumps, and secondary expansion tanks, a helium blanket is provided. Helium was chosen as the gas blanket since it does not react with the molten sodium, react with the surface of the stainless steel pipe and equipment (as would be the case with nitrogen), or become radioactive (as would argon).

The helium is supplied from a number of high-pressure helium cylinders manifolded to a pressure-reducing system and distributed throughout the plant to the various pieces of equipment requiring a helium blanket.

There are two nitrogen cooling systems: one a high-pressure system for cooling the reactor top shield and one which provides an inert gas atmosphere around the primary sodium piping and equipment. The latter system contains water-cooled nitrogen coolers located in accessible areas through which the nitrogen gas is recirculated. Nitrogen will be supplied to the plant from tank trailers through a reducing and storage station and will be distributed throughout the plant.

The objective of the radioactive vent system is to collect and monitor all radioactive gases for direct release to atmosphere, if the activity is below the maximum permissible concentration (MPC), or for storage in decay tanks and controlled release to the atmosphere.

If the pressure at the source of the various contributing systems and components connected to the radioactive vent system increases beyond the control valve or relief valve settings, the valves open, admitting gas to the vent system. Gases from the vapor traps and reactor atmosphere flow directly to the suction tank. All other gases flow to a header and past one of the radiation monitors. If the activity level of the gas is below the MPC, the radiation monitor will open the valve on the line that discharges to the effluent stack.

If vent gas is too radioactive, it is compressed, stored in holdup tanks at 150 psi, and, after sufficient time has been allowed for the radioactivity to decay, discharged by controlled release to the stack.

A reactor auxiliary power system provides an electrical distribution system that will meet the HNPF requirements for the following electrical loads:

Startup Loads

These loads include pumps, heaters, and other loads necessary to start the plant.

Reactor Auxiliary Loads

These include pumps, fans, compressors, and other loads essential to the continuous operation of the reactor.

General Service Loads

These include lights, heating and ventilation receptacles, cranes, etc., required for maintenance, comfort, and operation.

Vital Service Loads

These loads include pumps, battery chargers, lights, vital heating and ventilation, fuel cask gantry, and related equipment that can tolerate only a short time outage for protection of equipment and personnel.

Critical Loads

These loads require an uninterrupted source of power to protect personnel and equipment.

Primary power for the reactor electrical loads will be 4160 volts, three-phase, three-wire, resistance grounded wye, 60 cycles per second. This primary power will be available from the normal station service transformer and the reserve station service transformer. This power will feed the reactor primary 4160-volt bus. The full load electrical demand for the reactor will be approximately 3500 kva.

A radioactive liquid waste system is provided primarily for handling the contaminated water from the fuel wash cells. The fuel elements as they come from the reactor have sodium clinging to them which must be removed by washing with steam and water before storage and shipment. The wash water is a dilute solution of radioactive sodium hydroxide.

This wash water is stored in decay tanks. If the solution contains only short-life radioactive material, it is eventually dumped to a leaching field. If the solution contains long-life radioactive material, it is concentrated, packaged, and shipped from the plant site.

A cooling water system is provided supplying water from a cooling tower to various heat exchangers in the plant such as the loading-face-shield coolant exchanger, the spent-fuel pit exchanger, and the nitrogen coolers.

The reactor cavity liner and the spent-fuel storage pits have a separate coolant system containing buffered demineralized water. The exchanger for this system is cooled by cooling tower water.

Instrument air and service air are supplied to the nuclear plant facility by compressors located in the conventional plant.

Alarm signals are employed throughout the plant to call the attention of operating personnel to conditions that are abnormal in nature and require corrective action in order to protect the integrity of the plant. The alarm signals for these conditions will be distinctive so that they can be easily distinguished from all other plant noises and each will be easily identified with the type of condition it is signaling.

The communicating system is quite extensive and highly reliable in order to adequately cover the relatively complex heat transfer plant and to insure the proper operation of the nuclear facility. It is of paramount importance that the communications system be dependable in case of any irregularities in the operation of the nuclear facility in order to take proper corrective action to insure the safety of the plant and plant personnel.

SPECIAL DESIGN FEATURES

An unusual feature of sodium-cooled reactors is the requirement of a heating system for all sodium components. Since sodium is solid below 208°F, it must be preheated to a higher temperature. In addition, all other components in the sodium system must be uniformly preheated to the sodium temperature to prevent sodium solidification and to minimize thermal stresses. The preheating temperature selected for the HNPF is 350°F. The heating system consists of electrical tubular heaters located in annular chambers around the heated equipment. The chambers are made of stainless-steel sheet, which also supports the thermal insulation. The temperature in the heated chamber is measured by thermocouples which are connected to scanning controllers. The entire heating system requires over 1100 heater sections and control points. The total preheat load is approximately 500 kw.

The entire sodium system is stainless steel. Because of the relatively high coefficient of expansion, low thermal conductivity, and high operating temperatures, growth and stress problems are amplified. The behavior of the entire system was investigated under thermal shock conditions. The design of critical components are of unique configuration with thin walls to accommodate rapid temperature changes.

The steam generators consist of duplex tubes with an elaborate pressure relief system to relieve pressure which would be built up by the exothermic reaction resulting from the mixture of sodium and water in the unlikely event of a double tube rupture.

The intermediate heat exchangers (IHX) are preheated from the outer shell in a controlled manner to avoid buckling or distortion of the outer tubes or separation of the tube connections. The IHX, when brought up to system temperature by preheat, can receive the hot sodium without damaging stresses. Due to the bellows provided with the IHX, a unique support arrangement was required to maintain alignment of the two sections and avoid torsional and bending loads.

The reactor top shield consists of a matrix of holes which has to support the dead weight of the shield, support the handling equipment loads, the suspended control and shim rods, and withstand the shock loads of safety rods during a reactor scram. The top shield separates the core temperature of about 1050 °F from the room temperature of usually 75°F. Cooling pipes at the lower surface of the shield keep that surface at about 150°F.

Support of the reactor vessel is unique since a firm and level support structure is provided for the containment vessel which in turn supports the reactor vessel, thermal shields, and insulation. The support structure is in the form of many short pipes, standing on end as columns with a combination shim and cap on the top. The cap will be machined to provide any shimming required for leveling. The pedestals are welded to the cavity liner. The pipes are 18 inches in outside diameter. The columns plus the shim caps have a nominal height of 25 inches.

PLANT CONTROL SYSTEM

During steady-state operation, steam pressure and temperature at the turbine are constant; the reactor outlet temperature is constant; the reactor inlet temperature is programmed to decrease with decreasing primary flow; flux is adjusted so that thermal power produced by the reactor balances electrical power output plus heat losses; secondary hot-leg temperatures are not controlled and increase with decreasing load; secondary cold-leg temperatures are not controlled and decrease with decreasing load; and evaporator temperature and pressure and feedwater temperature decrease with decreasing load.

Under these conditions, the principal forcing functions are steam flow, temperature, pressure, and feedwater temperature. The quantities directly being varied are sodium flow (pumps) to maintain heat transfer, neutron flux (control rods) to maintain constant reactor outlet, and feedwater flow (valves with constant upstream pressure control by others) to maintain evaporator level. The purpose of the plant control system on load-following is to establish steady-state conditions when the demanded power is at a constant value.

An increase in load is established by opening the turbine throttle, which will also cause a drop in steam pressure at the header and, subsequently, further upstream.

An increase of reactor power and sodium flows must be accomplished to achieve steady-state at the new (increased) power level.

The increased steam flow and the decreased pressure will both present an increased power demand to the control system. The demand will cause an

increase in sodium flow demand in all circuits (according to the circuit flow ratios set by the operator and steam-to-sodium relations in the division of load computer).

As an increase in demand sodium flow is presented to the secondary flow controller, a flow error is created, causing an increase in demand pump speed. The speed control loop will cause the current in the eddy current coupling to be increased, which will increase flow.

A simultaneous flow demand (increase) signal will be given to the primary controller, causing an increase in flow as in the secondary loops above. When the flow actually changes, the demand value of reactor inlet temperature will also be increased. When demand temperature exceeds the measured value, a temperature error will be created, calling for a trim correction on the primary flow. This value can be a slight increase or decrease, depending upon heat performance and control schedule.

As the sodium flow in the primary loop increases, the demand neutron flux established by the nuclear power computer will rise proportionally. If the process were to continue with control action, the core outlet temperature and later the reactor outlet (plenum) temperature would decrease, creating a further increase in demand flux. As the demand flux increases and the actual flux does not, an error will be presented to the neutron flux controller, which actuates the rod control drives (driving the rods out). This action will continue until the reactor power rises to the new power level being demanded.

As steam temperature falls and steam flow increases, attemperation duty increases (below about 50%) or decreases (above 50%). In either case, a decrease in attemperated steam temperature creates a decrease in demand attemperator valve position (superheater bypass flow) which calls for the valve to move in the closed direction. At less than 50% power (provided the outlet temperature were to rise, and calculations say that it should) the valve would later move toward the open position as valve position demand increased.

As steam flow increases with no control action, the feedwater level would first increase due to swelling and then decrease as the actual level falls. This would create an increase in demand feedwater valve position. Also, feedwater flow less than steam flow (including blowdown) creates an error calling for

increased feedwater valve position. The increase in demand valve position calls for the valve to move to the open position.

All the aforementioned actions would take place in two steps. The first step would be an increase in thermal power to slightly more than the new steady-state value in order to compensate for the steam pressure drop. The second step would be a more gradual reduction as steam pressure increased to design steady-state.

In the load decrease situation, the results and control actions are exactly in the opposite sense in each case, with the following distinction: As steam pressure rises, it may rise (under rapid load changes) momentarily above the set value of steam dump pressure. In this case, a pressure error would be generated in the steam dump controller which would call for opening the steam dump valve. When the pressure setting was reduced below the set pressure for dumping, the valve would close.

Variable orifices are used in lieu of orifice plates for each fuel element to approximate an isothermal condition at the core outlet. The variable orifice permits removal of heat energy at a higher temperature level and permits adjustment during reactor operation. Adjustments are made from the reactor floor by using a dual functioning socket type wrench which provides means of rotating the variable orifice drive screw and a depth gage to indicate the elevation of the variable orifice plug.

ADVANTAGES OF SODIUM GRAPHITE REACTOR SYSTEMS

The Sodium Graphite Reactor (SGR) is ideally suited for power plants because it has the potential of producing steam conditions which are typical of modern conventional steam plants. Current development work and the designs which are evolving indicate the capability of SGR nuclear steam power plants to produce steam at 2400 psig, 1050°F primary steam conditions with 1000°F re-heat in the very near future, and with the potential of exceeding the current temperature and pressure capabilities of the steam plant.

Sodium heat removal characteristics are excellent and permit high specific powers. Leak-tight systems are essential for the primary system on all reactors, but it is especially important for sodium reactors because of the chemical reactivity of sodium, its high induced radioactivity, and the possibility of severe corrosion if sodium is not kept free of oxygen.

Sodium coolant exhibits the following features:

- a) Temperatures comparable to temperatures found in modern steam power plants are attainable and thermal efficiency is high.
- b) A wide range of coolant temperatures are available. Because of the large temperature rise in the coolant the flow is relatively low and requires low pumping power. However, the high difference in temperature across the reactor and heat exchange equipment introduces the possibility of high stress during thermal transients. There is also a stratification tendency, and thermal eddy currents may be induced in the liquid. These factors can be minimized by careful design.
- c) The low operating pressure permits thin wall construction, reduces rupture hazard, and facilitates rapid fuel handling.
- d) Physical and chemical characteristics include a high boiling point (1620°F) and a high melting point (208°F).

The graphite moderator features can be summarized as follows:

- a) Such systems possess high temperature stability, low vapor pressure, and low neutron cross section with long slowing down and diffusion lengths, permitting fuel elements to be widely spaced for easy

handling. However, the reactor core must be physically large to achieve adequate neutron economy.

- b) Mechanical and chemical characteristics include good structural properties at high temperatures, excellent machining characteristics, and chemical inertness to most materials. However, it must be physically separated from sodium in the reactor, which is accomplished by using stainless-steel skin material.

The combined features of sodium coolant and graphite moderator results in a reactor with particularly advantageous safety features. It is noteworthy that the Hallam SGR is the only nuclear power plant under construction in the U.S.A. which is housed in a conventional building.

The Sodium Graphite Reactor System provides the potential for building nuclear power plants having high thermal efficiency (above 40% net), very high specific power, low fuel cycle costs, low unit plant costs, and no obvious limit on plant size. Recent studies indicate that Sodium Graphite Reactor electrical plants can be constructed with capital costs that compare favorably with all other current reactor types. Studies of 250,000 ekw and 300,000 ekw nominal have resulted in net plant thermal efficiencies of about 40% and above and installed prices of below \$200 per net ekw.