K. W. Foster 10/15/64

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1000-MWE SGR
AND PROTOTYPE EVALUATION STUDY
VOLUME 2 - 200-MWE SGR
PROTOTYPE PRELIMINARY DESIGN

AEC Research and Development Report

ATOMICS INTERNATIONAL

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Prototype of SGR Nuclear Power Station

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FOREWORD

This document, the second of two volumes, describes a 200-Mwe high performance nuclear power plant which utilizes a sodium cooled, graphite moderated reactor (SGR) as the energy source. The plant is designed to serve as a prototype for very large advanced SGR systems in the 500- to 1500-Mwe range, which will fulfill the needs for low cost power in the 1970 era. Volume I reviews the philosophy supporting the development of large SGR power stations, their associated economic and technical advantages, and describes a conceptual design of a 1000-Mwe plant.

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I. SUMMARY

This document presents the preliminary design of a 200-Mwe prototype plant of an advanced 1000-Mwe sodium graphite reactor power station. The prototype is the next logical step in the progression to larger plants of this type. The plant incorporates many features of SGR power stations in the range of 500-1500 Mwe. The purpose of the prototype is to demonstrate the present capability to design, construct and operate a large SGR of improved design and performance characteristics and to provide operating experience with components required for later SGR plants with higher capacity and more advanced performance capabilities.

The 200-Mwe prototype plant represents major advances in SGR technology, and is based on construction and operating experience with the Sodium Reactor Experiment (SRE) and the Hallam Nuclear Power Facility (HNPF) and on current AEC development programs in the areas of fuel, core structure and components, fuel handling, steam generator and other related plant components. The unit is designed for construction at an average undeveloped site in the United States and is intended to serve as a load-following, self-sufficient power plant in a representative utility system. A sodium-cooled, graphite-moderated, uranium carbide fueled reactor system supplies the required steam flow at 2400 psig and 1000°F with a single reheat to 1000°F, to an indoor turbine-generator. A net station efficiency of 41.8% is estimated for these steam conditions. Rated gross electrical output of the plant is 210,000 kwe; ultimate gross capability is expected to be 221,000 kwe. Major design features of the prototype plant are as follows.

- a) Calandria core, which permits operation at high sodium temperatures and high core power density, and provides improved neutron economy.
- b) Uranium carbide fuel, which is capable of high temperature, high power density operation with extended burnup, leading to low fuel cycle costs.
- c) Once-through, modular steam generator design, which affords low fabrication costs and increased reliability at high steam conditions.

The calandria is a single, large vessel which contains the graphite moderator. Process tubes penetrate through the graphite and are brazed to the top and bottom heads of the calandria. The brazed joints, readily melted and rebrazed by remote induction heating techniques, assure the maintainability of the thin-wall process tubes and the graphite. Austenitic stainless steel is used for the calandria structure.

The nuclear fuel material is hypostoichiometric uranium carbide, clad in stainless steel and assembled into multirod clusters. The fuel element is designed to operate with a fuel temperature of 1750°F, resulting in a peak specific power of 52 kw/kg U. The average fuel exposure is 25,000 MWD/Mtu.

The steam generators consist of high-temperature stainless steel superheater modules and low-temperature chrome-moly steel evaporator modules. The modules are vertical, once-through design with sodium on the shell side and steam on the tube side. The utilization of the module concept provides a reliable, easily maintainable, low cost steam generator.

The power plant is a self-sustaining unit with all auxiliary facilities and essential services provided as required. The major structures consist of the turbine generator building and the adjoining reactor building. The reactor building consists of a high bay over the reactor and fuel handling facilities, and a low bay for housing the primary sodium pumps and intermediate heat exchangers. Both bays are of structural steel framing and are enclosed by precast concrete panels. The turbine generator building is a one-story steel frame structure with insulated metal siding, and a reinforced concrete slab forming the operating floor. Additional buildings are provided for administration, warehouses, machine shop and radioactive waste storage. Facilities in the latter are provided for the processing and disposal of all liquid, gaseous, and solid radioactive waste.

The reactor is fueled with uranium carbide, cooled by liquid sodium and moderated and reflected by graphite. Rated core output is 480 Mwt. The reactor is housed in a steel-lined concrete cavity below the operating floor level of the building. Beginning at the outer perimeter of the concrete cavity and moving radially inward, the reactor structure consists of a cavity liner, thermal shield, reactor vessel and its external liner, and the calandria vessel containing the core internals. The calandria vessel contains the reflector and

moderator graphite and is penetrated by stainless steel process channels which provide a flow path through the core for the sodium coolant. Within the process channels are located the 168 fuel elements and 24 control elements. The active core forms a right circular cylinder 13.5 ft in diameter and 14 ft high, and contains 20,830 kg of uranium.

Heat is removed from the reactor core and delivered to steam generating facilities by the sodium heat transfer system which consists of a main circuit of 432 Mwt (90%) capacity, and an auxiliary circuit of 48 Mwt (10%) capacity. Each circuit consists of a primary and secondary sodium loop, intermediate heat exchangers and steam generating facilities. The primary sodium flows through the core and then through a sodium-to-sodium intermediate heat exchanger where the heat assimilated in passing through the core is transferred to secondary sodium. The primary sodium becomes radioactive in passing through the core, and thus a secondary sodium loop is employed to transport the heat to the steam generator. The secondary sodium does not become radioactive, thus allowing access to the steam generating facilities during operation. The primary sodium loops, including the intermediate heat exchangers, are enclosed by concrete shields. The sodium pumps are vertical, centrifugal, free-surface units of 30,700 gpm capacity in the main circuit and 3400 gpm in the auxiliary circuit. The steam generator consists of high temperature superheater modules, low temperature evaporator modules and reheater modules, of once-through design.

The rated turbine generator capability is 210,049 kw, with steam conditions of 2400 psig/1000° F/1000° F, a turbine back-pressure of 1-1/2 in. Hg abs., and 0% makeup. The turbine is a two-casing machine with main steam and reheat steam entering the high pressure intermediate pressure casing. The HP steam exhausts to the reheater and the IP steam exhausts to the double-flow low pressure turbine. There are seven extraction states for feedwater preheating. Feedwater is delivered to the steam generator by two half-capacity pumps driven by electric motors through fluid coupling. A 10% capacity steam-driven pump is also provided for startup and shutdown purposes. The main condenser is a 68,500-sq-ft, 1-pass, divided water box unit designed for 1.5 in. Hg absolute pressure at turbine rating. The generator is a hydrogen-cooled, 3600-rpm unit rated at 256,000 kva, at 45 psig H₂ pressure. The voltage generated by the conventional type electrical system is 22 kv, which in turn is stepped up to the

138-kv transmission voltage by a 250-mva main transformer. Emergency power systems are provided to serve loads which require power continuity.

Station control is designed to permit automatic load-following operation over a range of 20 to 110% of rated load, with semiautomatic operation of the plant at any load down to no-load, hot standby condition. Core power output is changed according to the turbine generator load demand by varying the sodium flow rate while maintaining the reactor delta T approximately constant. The change in primary sodium flow then calls for a compensating change in nuclear heat rate by control rod movement. Reactor system response time to load changes is shorter than for the conventional part of the plant, and therefore is not limiting. The central control room is located at the turbine operating floor level. Master station controls are located there. Local control panels are provided outside the central control room for certain process systems.

Nuclear instrumentation for power operation is located in thimbles outside the reactor cavity liner. Incore startup nuclear instrumentation is placed within thimbles located in the process channels.

Fuel element transfer and movement of other core components are performed with a fuel handling machine carried on a gantry crane. The machine consists of component handling mechanisms, a control console, lead shielding, and support equipment to cool a spent fuel element and maintain an inert gas atmosphere in the machine. Storage facilities are provided for both spent and new fuel.

The fuel management program is based on an equilibrium cycle average fuel burnup of 25,000 Mwd/MTU. Refueling of 1/6 of the core is required every 6 to 7 months. Shutdown time required for a refueling is approximately three days.

The major design and performance characteristics of the plant are listed in Table 1.

TABLE 1
PLANT DESIGN AND PERFORMANCE CHARACTERISTICS

Steam conditions	
Main steam pressure (psig)	2400
Main steam temperature (°FTT)	1000
Reheat steam temperature (°FTT)	1000
Turbine-generator	
Rated gross electrical output (kwe)	210,049
Ultimate gross electrical output (kwe)	221,000
Net plant heat rate (Btu/kwh)	8160
Net plant efficiency (%)	41.8
Type of generator	3600 rpm, 2-pole H ₂ -cooled
Type of turbine	TCDF-30 in
Main steam flow (lb/hr)	1,351,700
Reheat steam flow (lb/hr)	1,183,800
Condenser pressure (in Hg abs)	1.5
No. of feedwater heaters	7
Final feedwater temperature (°F)	473
Nuclear steam generating system	
Reactor	
Core thermal power (Mwt)	480
Fuel	uranium carbide
Active core size (ft diameter x ht)	13.5 x 14
Number of fuel elements	168
Core loading (kg U)	20,830
Lattice spacing (in) square	10.25
No. of control elements	24
Enrichment (wt% U ²³⁵)	3.80
Average fuel burnup (Mwd/MTU)	25,000
Average specific power (kw/kg U)	23.0
Fuel elements	
Geometry	18-rod cluster
Fuel slug diameter (in)	0.50
Active length (ft)	14
Maximum hot channel fuel temperature (°F)	1750
Maximum linear rod power (kw/ft)	24.0
Sodium heat transfer system	
Main circuit	
Thermal capacity (Mwt)	432
No. of loops	1 primary, 1 secondary
No. of intermediate heat exchangers	2
No. of steam generator banks	2
Primary Loop	
Sodium temperature (°F)	
Reactor inlet	750
Reactor outlet	1150
Sodium flow rate (lb/hr)	12.2 x 10 ⁶
Secondary Loop	
Sodium temperature (°F)	
Steam generator inlet	1075
Steam generator outlet	675
Sodium flowrate (lb/hr)	12.2 x 10 ⁶
Auxiliary	
Thermal capacity (Mwt)	48
No. of loops	1 primary, 1 secondary
No. of intermediate heat exchangers	1
No. of steam generator banks	1
Primary Loop	
Sodium temperature (°F)	
Reactor inlet	750
Reactor outlet	1150
Sodium flowrate (lb/hr)	1.35 x 10 ⁶
Secondary Loop	
Sodium temperature (°F)	
Steam generator inlet	1075
Steam generator outlet	675
Sodium flowrate (lb/hr)	1.35 x 10 ⁶

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II. SITE AND PLANT ARRANGEMENT

A. SITE DESCRIPTION

Since an actual site has not been established for the power plant, it is assumed to be constructed on the hypothetical reference site¹ as defined by the AEC. A brief description of this site is outlined below.

1. Location

The site is located on the east bank of the North River, 35 mi north of Middletown in an area of grass-covered, level terrain. It is .25 ft above minimum river level and 5 ft above maximum river level.

2. Access

Highway access is provided by an existing secondary road which connects to a State Highway 15 mi east of the site. The road is in good condition. A main line of the B & M Railroad is 5 mi from the site. The river is navigable throughout the year for boats with up to 6-ft draft. An airfield is located 3 mi from the State Highway and 15 mi from Middletown.

3. Utilities

The river provides an adequate source of raw makeup and condenser cooling water. The average maximum temperature is 75°F, and the average minimum is 40°F. Natural gas service is available 4 mi from the site boundary. Construction power is available at the southeast corner of the boundary.

B. GROUND IMPROVEMENTS

The site is graded and provided with culverts and drains as required. Security fencing is erected to enclose the plant area and switchyard. Appropriate paved access roads, parking, yard lighting and landscaping are provided. A spur track is constructed from the B & M main line to the site and inplant railroad trackage is constructed for service to the reactor and turbine buildings and the switchyard. A barge landing facility is also provided.

1. Guide to Nuclear Cost Evaluation, TID-7025, Kaiser Engineers

Yard piping systems and facilities include a plant fire protection loop with hydrants and hose houses; service and potable water distribution systems; a sanitary sewage collection and discharge system including sewage treatment facilities; and all other auxiliary systems required for a completely integrated nuclear power plant.

C. PLANT ARRANGEMENT

The plant is located on the east bank of the river which runs through the site. The Plot Plan (Drawing 1) shows the arrangement of the various plant buildings and facilities on the overall plant site. The main buildings include the reactor building, turbine-generator building, and service buildings for administration, warehousing, shops, and waste processing.

The reactor building encloses the reactor, the fuel handling system, the primary heat transfer systems and auxiliary service systems. Attached to the east end of the building is the radioactive waste facility, which includes the building exhaust ventilation system and equipment decontamination areas as well as the liquid and gaseous radioactive waste systems. The radioactive gas decay tanks are located on a concrete pad adjacent to the radioactive waste facility. These tanks are surrounded by an exclusion fence.

Attached to the west end of the reactor building is the steam generator enclosure. Near the top of this structure is the secondary pump service room, which is the only area in this enclosure that is occupied during the performance of routine operations. Other areas need be entered only for special servicing or maintenance inspections. The control room is next to the steam generator structure and connected to the reactor building by the cell cooling and a service area.

The turbine building is located west of the control room for economic tie-in to the steam generators, and for operating convenience. The building is located adjacent to the river, and the condenser in the turbine generator building is located directly opposite the intake structure. The main transformer and the switchyard face the outgoing transmission line to the east. The switchgear room is located at the east end of the basement since a majority of the heavy electrical loads are in the turbine area, and also because the transformers are located outside the north wall of the building close to the switchgear. A machine shop, an auxiliary 150-psig steam boiler, and

water-treating equipment for makeup and condensate scavenging are located at the south end of the building. The administration building, warehouse, and machine shop face the access road to the north. Normal access to all structures is provided through the administration building. Chemical laboratory, locker rooms, showers, and monitoring station are located in the radioactive waste processing area.

Two parking lots for 40 cars each are located in the southeast and southwest corners of the lot.

The plant arrangement is planned for future expansion. A second plant can be added to the north as indicated on the Plot Plan. The cost of the second plant would be reduced because it would utilize or add only a small additional capacity to the following equipment or systems: (a) turbine gantry crane, (b) fuel handling machine, (c) maintenance cell and equipment, (d) fuel handling system, (e) control room, (f) R/A waste facilities, (g) health physics facilities, and (h) decontamination facilities.

The plant arrangement is further illustrated on the following: Drawing 2. Reactor Building, Plan View at elevation 112 ft; Drawing 3. Reactor Building, Plan View at elevation 95 ft; Drawing 4. Reactor Building, Plan View at elevation 75 ft; Drawing 5. Reactor Building, Section A-A; Drawing 6. Reactor Building, Section B-B; Drawing 7. Reactor Building, Section C-C; Drawing 8. Turbine Building, Plans; and Drawing 9. Turbine Building Sections.

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III. STRUCTURES

A. SUBSTRUCTURES

The reactor operating deck and the steam generator operating deck are set at 100-ft elevation, with the surrounding site grade at 70-ft elevation. Thus, the bulk of the excavation will be in the Brassfield limestone with only a small area around the reactor penetrating approximately 6 ft into the Richmond limestone (deepest excavation is approximately 36 ft). The soil conditions permit the use of slab footings as indicated on the plant arrangement drawings.

Below-grade basement areas are protected from groundwater inleakage. Multilayer waterproofing membrane is applied to all basement walls and slabs. In addition, a cast-in-place drain system is provided within the walls of the deep area around the reactor.

B. TURBINE-GENERATOR BUILDING

The turbine-generator building is a structural steel framing, insulated sheet metal siding structure supported by single footings on rock. It encloses the turbine generator operating deck and the auxiliary equipment on the lower levels. A 50-ton building crane services the operating deck.

The turbine pedestal, located within the building, is a reinforced concrete structure. It is built on a 5-ft concrete mat on rock and separated from the building by a continuous vibration joint. The pedestal supports the condenser at the ground floor level and the turbine-generator and exciter at the operating level.

All electrical gear, pumps, compressors, and other secondary items are located on the ground floor. The main and auxiliary transformers are adjacent to the building within the switchyard fence.

The turbine-generator building is adjacent to the control room and steam generator areas in the reactor building so as to minimize the steam-line runs. A freight-passenger elevator provides access to the steam generator structure as well as the turbine-generator building.

C. REACTOR BUILDING

The reactor building, similar to that at Hallam, is designed as a controlled leakage structure with a slightly negative internal pressure maintained within the building by the ventilating system. Special design features include (a) leak-tight construction with a leakage rate no greater than 145% of the building volume per day at 0.25 in. of water pressure differential; (b) the use of steel plate to line cells containing primary sodium system equipment, pipeways, and service cells; and (c) the use of sealed penetrations through all leak-tight enclosures for personnel and equipment access or for piping, electrical, and instrument connections.

The superstructure is of standard "tilt-up" construction. The concrete wall panels are approximately 6 in. thick and are attached to the columns with nuts which are fastened to Nelson studs extending through the wall panel to the columns. The bolt holes are oversized to allow for thermal expansion and movement of the panels. To prevent air leakage all panel joints, holes, and other openings are continuously calked.

The roof is a multilayer hot-mopped composition roof placed over metal paneling similar to Robertson paneling. Again all seams or joints are calked or welded. Sheet-metal flashing is provided at the junction between the walls and ceilings, and at all other points where necessary to stop air leakage. A calked joint is provided at the junction between the wall panels and the substructure concrete. All caulking is done utilizing thiokol or thiokol-epoxy combinations.

The primary sodium systems cells include the intermediate heat exchanger cells, the fill tank cell, the service cells, the pipeway, and the reactor cavity. Seven-ft-thick concrete walls surround these areas where shielding is required. The cells are lined with 1/4-in.-thick carbon steel liner plates. The liner plates provide a means of sealing these cells gas-tight so that a nitrogen atmosphere can be maintained within the cells.

All concrete walls are constructed of normal concrete obtainable within the local area. Dense concrete (245 lb/ft^3) is required only in the maintenance cell operating face, the top of the reactor and fuel storage cells, and over the valve area in the IHX cells.

Two cranes are provided in the reactor building. In the high-bay area an 80-ton crane runs the length of the building and is supported from the building columns. Similarly, a 25-ton crane is provided in the low-bay area and is supported by columns or the elevated IHX cell concrete wall.

No windows are used in the building. The large equipment access doors are sliding steel doors with inflatable seals. Personnel access is through metal doors set in vestibules with weather-stripping around each door.

D. CONTROL ROOM AND SERVICE AREA

The control room and service area is a 4-story structure consisting of structural steel framing with reinforced concrete floors. It is supported by single footings on rock and enclosed by insulated metal siding. It contains the personnel control facilities, the health physics and radio chemistry offices and laboratories as well as the control room which overlooks the turbine-generator areas on the operating floor level. A lower level contains locker and lunch rooms and the cable pulling room. The switchgear room and the instrument repair and calibration shop are located on the ground floor. The emergency diesel generator and the battery room are also located in this area.

E. STEAM GENERATOR AREA

The steam generator area is a sheet-metal siding structure supported by structural steel framing. It is approximately 130 ft high. The structure houses the steam generators, the secondary sodium pumps, and the blowdown tanks. A concrete slab is provided at 178-ft elevation to support the dynamic load of the pumps.

The steam generator structural steel is supported on the concrete basement walls. A collection sump and pump are located at the low point of this substructure to remove any water which may enter the area.

F. RADIOACTIVE WASTE AREA

The radioactive waste processing area will be a 4-story structure. The below-grade area is cast-in-place reinforced concrete structure. The above-grade portion is constructed in the same manner as the reactor building. The same leak-tightness criteria are maintained for this structure.

The ventilation stack is a mild steel stack supported against the reactor building and extending approximately 20 ft above the roof.

G. ADMINISTRATION BUILDING

The administration building, located near the main gate, is a 30 x 50 ft one-story structural steel framing. The building is supported by single footings on firm undisturbed earth. Office areas are enclosed by curtain-type walls with metal mullions and panels of glass or insulated porcelain enameled steel.

The building contains the plant administration office, private offices for supervising personnel, and a conference room; and is air-conditioned.

H. WAREHOUSE AND MACHINE SHOP

The warehouse and machine shop is 50 ft x 100 ft and located southeast of the turbine-generator building. The shop is a standard, commercially available, structural steel, insulated metal siding building similar to a "Butler" building. The building contains a warehouse, storeroom area, and a machine shop and is serviced by a 10-ton traveling bridge-crane. Major equipment includes a 15-in. lathe, a milling machine, a drill press, and a welding booth.

The building is provided with thermostatically controlled unit heaters, propeller fans, and manually operated louvers.

I. SURFACE COVERING AND PROTECTION

Exposed surfaces will be finish-painted, with the exception of concrete, nonferrous metal, galvanized steel, and factory-finished surfaces or equipment. Roof ducts, vents and piping above roof will be painted with aluminum paint. The inside of the maintenance cell is coated with phenoline-type radiation-resistant paint. All surfaces in the maintenance cell operating and service areas will be covered with hard-finish enamel paint.

IV. REACTOR

The reactor is fueled with uranium carbide, cooled by liquid sodium, and moderated and reflected by graphite, thereby operating in a thermal neutron spectrum. The rated core output is 480 Mwt. The reactor is housed in a steel-lined concrete cavity in the operating floor of the reactor building. The cavity is capped with the loading face shield, consisting of 6 ft of dense concrete and reinforcing steel, which forms a continuation of the operating floor over the reactor and provides radiation protection for operating personnel.

Beginning at the outer perimeter of the concrete cavity and moving radially inward, the reactor structure consists of: (a) cavity liner, (b) thermal shield, (c) reactor vessel and its external liner, and (d) calandria vessel and internals. A simplified diagram of the reactor structure is given in Figure 1. Drawings 10 and 11 provide further details. Each component of the reactor structure and its function are described in more detail in Section IV-A.

The calandria vessel contains the reflector and moderator graphite and is penetrated by stainless steel process channels which provide a flow path through the core for the sodium coolant. Within the process channels are located the fuel elements and control elements. Above the calandria vessel are the upper sodium plenum, an inert gas blanket, reflective insulation, and the loading face shield; below it is the sodium inlet plenum.

The active core contains 168 fuel elements and 24 control elements, and forms a right cylinder 13.5 ft in diameter and 14 ft high. Initial fuel enrichment is 3.80 wt % U^{235} . The core is surrounded by a 2-ft-thick axial and radial graphite reflector. Design data for the core are tabulated in Table 2.

Primary sodium is brought to the reactor at 750°F through a double-walled pipe which enters the bottom of the reactor vessel. Sodium flows upward into the inlet plenum, then through the process channels and fuel elements, and into the upper plenum where it is removed from the reactor vessel through a single outlet line. Mixed mean sodium outlet temperature is 1150°F. From the reactor vessel it flows through a sodium-to-sodium intermediate heat exchanger (IHX), where the heat assimilated in passing through the core is transferred to secondary sodium. The primary sodium then returns to the reactor to repeat the cycle.

Figure 1. Reactor Structure

TABLE 2
REACTOR CORE

Thermal power rating (Mwt)	480
Active diameter (ft)	13.5
Active height (ft)	14
Fuel elements (number)	168
Control elements (number)	24
Lattice spacing — square (in.)	10.25
Core loading (kg U)	20,830

Power production in the core is varied by control rod movement according to the turbine-generator load demand. A change in heat removal requirements of the core is met by varying sodium flow rates while maintaining the reactor ΔT approximately constant. Reactor system response time to load changes is shorter than for the conventional part of the plant, and therefore is not limiting.

Nuclear instrumentation for power operation is located in thimbles outside the reactor cavity liner. Incore startup nuclear instrumentation is placed within thimbles located in the process channels. This instrumentation is temperature-limited and is used during initial startup and testing when the reactor coolant temperature does not exceed 600°F.

Fuel element transfer and movement of other core components are performed with a fuel-handling machine carried on a gantry crane which travels over the reactor on rails embedded in the reactor room floor. The machine consists of component handling mechanisms, a control console, lead shielding, and support equipment to cool a spent fuel element and maintain an inert gas atmosphere in the machine. Storage facilities are provided for both spent and new fuel.

The fuel management program is based on an equilibrium cycle average fuel burnup of 25,000 Mwd/MTU. Refueling of 1/6 of the core is required every 6 to 7 mo.

Shutdown time required for a refueling is approximately 3 days.

A. DESCRIPTION

1. Loading Face Shield (Drawing 12)

The stationary loading face shield is located above the reactor core and forms a cap over the reactor cavity. The total depth of the shield is 6 ft, 4 in. and it is penetrated by 308 sleeves and plugs for fuel handling and calandria maintenance. It is fabricated of structural steel and high-density concrete. The purposes of the shield are to: (a) Attenuate the radiation emanating from the reactor below, (b) Support the reactor vessel and the calandria, (c) Provide a thermal shield from the upper plenum sodium, (d) Serve as a loading face for fuel elements and accessory component transfers and for calandria maintenance, and (e) Provide a gas seal between the reactor vessel and the reactor building.

The holes through the loading face shield are on a 10.31-in.-square lattice. They are 9-in.-ID casings, stepped down to 8-in.-ID in the lower half of the shield. Suspended from the bottom surface of the shield is reflective thermal insulation, consisting of 18 horizontal stainless steel sheets 1/32 in. thick, spaced 0.8 in. apart. Directly above the reflective insulation, a 1-in.-thick carbon steel plate supports a 3-in. layer of lead. Cooling of the shield is accomplished by compressed nitrogen gas as described in Subsection G. The shield concrete is maintained below 150°F. Further details of the loading face shield are given in Drawings 13 and 14.

2. Cavity Liner (Drawing 15)

A 27.5-ft-ID cavity liner made of 1/4-in.-thick carbon steel lines the walls and the bottom of the concrete reactor cavity and extends to the gallery diaphragm seals. The concrete walls of the cavity provide biological shielding between the reactor and the equipment cells. The liner provides a gas-tight envelope between the reactor cavity and the outside concrete structures and shields. This envelope prevents moisture from entering the reactor cavity and confines the nitrogen gas atmosphere surrounding the reactor structure and vessels. The liner also serves as additional containment for possible leakage from the reactor vessel or adjacent piping. As the reactor vessel is also contained within the reactor vessel liner, the cavity liner becomes a third enclosure for the sodium, thus providing tertiary containment.

The cavity liner and adjacent concrete are heated by radiation attenuation and therefore must be cooled. Cooling is provided by nitrogen gas circulating through ducts fastened against the interior surface of the liner. The concrete is maintained below 150°F.

3. Thermal Shield (Drawing 15)

The purpose of the thermal shield is to protect the concrete biological shielding from possible damage by heat generated from absorption of radiation. The thermal shield is an annular carbon-steel plate with an inside diameter of 20.8 ft. It is 4 in. thick for the lower 30 ft, and 2 in. thick for the 12.5 ft upper course. The shield is located between the reactor vessel and cavity liner, and is supported by the reactor cavity floor.

The shield material is effective in absorbing gamma radiation and providing inelastic scattering neutrons. Both processes cause heat to be generated within the shield. To impede heat transfer from the thermal shield to the cavity liner and concrete, 1-ft-thick thermal insulation is attached to the exterior surface of the thermal shield. The shield is cooled by thermal radiation back to the reactor vessel.

4. Reactor Vessel (Drawing 16) and Reactor Vessel Liner (Drawing 17)

The reactor vessel provides a container for the calandria core and the sodium coolant. It is made of 1-in.-thick austenitic stainless steel and is 42.5 ft high with a 19.5-ft ID. The reactor vessel is supported from the loading face shield by hangers (Drawing 18) fastened to its upper edge.

One of two large nozzle openings into the vessel is for main sodium inlet; the other, for main sodium outlet. The inlet nozzle enters the vessel through the center of the bottom head; the outlet nozzle is on the side of the vessel 3-1/2 ft above the calandria top plate. A cylindrical outlet manifold, shown in Drawing 19, is provided around the upper plenum which uniformly distributes the outlet sodium flow. In addition, there are other smaller penetrations for fill, drain, and venting purposes. Other reactor vessel details are shown in Drawing 20.

Under normal operating conditions the reactor vessel contains sodium to a level about 13.5 ft above the calandria top plate. The lower plenum area operates at 750°F and the upper plenum at 1150°F. Maximum operating pressure in the vessel is the static head of sodium, which is about 15 psig.

The purpose of the reactor vessel external liner is to provide secondary containment for the sodium in the reactor vessel and thus maintain the sodium level above the core in the event of a leak in the reactor vessel. The liner is made of 1/4-in. stainless steel and is fastened and supported at the top of the reactor vessel. The space between the liner and vessel is 1/4 in. and is filled with helium. The liner is shaped around the nozzle openings and its extensions connect to the guard pipes. A line is provided to drain the space between the reactor vessel and the liner.

5. Calandria Vessel (Drawing 21) and Internals

The primary purpose of the calandria vessel is to provide support and containment for the graphite moderator and reflector. The calandria and the process channels provide the means for separating sodium and graphite, while utilizing a minimum of structural steel within the active portion of the core.

The calandria vessel is 18.5 ft in diameter, 24.5 ft in height, and has an integral elliptical lower head. It is constructed from 1-in.-thick type 304 stainless steel, and is supported by hanger rods fastened to the underside of the loading face shield. The top of the vessel is capped by a 1/2-in.-thick top plate which is also supported by hanger rods fastened to the loading face shield. A 1-in.-thick bottom plate is located at the intersection of the cylindrical vessel wall and elliptical lower head, and is supported and fastened to the lower head by vertical rods. Details of the calandria vessel are shown in Drawings 22 and 23.

The lower head and the bottom plate provide a pressure plenum for the inlet sodium coolant. Since the dynamic pressure is contained in this plenum, the reactor vessel is relieved of all but the static sodium pressure and the cover-gas pressure which is essentially atmospheric. Operating temperatures are 750°F on the bottom and 1150°F on the top plate.

The moderator and reflector utilize nuclear grade graphite logs. Three types of logs are used: (a) a process tube center log which is cylindrical in shape and has a 4-in.-diameter hole through its center to accommodate a process tube; (b) a reflector center log, also cylindrical in shape, which has a 2-in.-diameter hole through its center to permit a handling tool to reach its lower end; and (c) a filler log which is nearly square on three sides and cylindrical and concave on the remaining side to mate with a center log. The process tube center

log is made up from three types of vertical pieces, namely, an upper section (Drawing 24), a lower section (Drawing 25), and five center sections (Drawing 26). The reflector center log is also made up from a number of vertical sections as shown in Drawing 27. The filler log is comprised of the three vertical sections shown in Drawing 28, and is fabricated from the blank graphite log illustrated in Drawing 29.

The graphite unit cell which is repeated throughout the core is made up of four filler logs and either a process tube center log or a reflector center log. The process tube center log is assembled with and as part of the process tube. Details on the calandria assembly are shown in Drawing 30.

The five graphite logs needed to make up a unit cell can be removed and replaced through the loading face shield hole associated with the cell. The replacement of process tubes and graphite is not expected to be a periodic maintenance item; however, the design provides for such maintenance.

Within the calandria vessel, the space between the graphite logs is filled with helium. The helium is used to provide a nonradioactive medium in which sodium vapor can be detected to indicate possible sodium leaks into the calandria vessel. The helium is circulated to an externally located sodium vapor detector. The circulating loop is tied into the plant helium supply system and the radioactive vent.

There are 192 process channels installed in penetrations through the moderator graphite, and 116 braze-joint sealed access holes in the reflector region. The process channels house the fuel elements and control elements, and provide a flow channel for the sodium coolant through the core. The process channel is divided into three functional parts: (a) the process tube through the core; (b) the orifice tube, which extends down from the loading face shield into the process tube; and (c) the fuel shield plug and orifice casting which extends down from the loading face shield into the orifice tube, and provides flow control. Figure 2 is a simplified sketch of the process channel showing the relationship between the functional parts. Drawings 31, 32, and 33 provide further details on each part.

The process tube is a 4-in.-OD by 0.025-in.-wall-type 304 stainless steel structure. It is installed with brazed joints at both ends to allow remote removal and reinstallation of process tubes and graphite. The calandria vessel

Figure 2. Simplified Process Channel

is designed so that any individual process tube and associated graphite can be removed and replaced. Details on the top and bottom brazed joints between the process tube and calandria vessel are shown in Drawings 34, 35, 36, and 37. Each process tube has a bellows (Drawing 38) near the calandria top plate just below the brazed joint to accommodate differential expansion between tubes and calandria shell, and between tubes operating at slightly different temperatures. The top of the process tube also incorporates a guide tube (Drawing 39) which protects the bellows from stress caused by the sodium flow or external lateral loads.

Flow through the fuel process channels is controlled by a variable orifice assembly, manually adjusted at the operating floor level of the loading face shield. An orifice tube rests on the step of the opening through the loading face shield and extends down to make a sliding fit into the process tube below. This tube is the stationary member of the variable orifice and contains two opposing 1/2-in. vertical slots 40 in. above the calandria top plate. These slots face triangular openings in the orifice casting extension of the fuel shield plug assembly, which rotates inside the orifice tube to produce an adjustable flow area through which the sodium coolant discharges from the process channel into the reactor upper plenum. The rotating fuel shield plug and orifice casting are fastened together below the loading face shield and function as one piece. This component is the adjustable portion of the variable orifice.

Coolant flow adjustments are made at the operating floor by a rotating device designed specifically to provide a vernier-type control over the range of rotation of the plugs. Adjustments are necessary only as required by changes in core power distribution.

The fuel shield plugs are equipped with quad-ring seals to provide a gas-tight seal to the top shield. There are two rings per plug. The second is for backup purposes, and both are replaceable. Reflector shield plugs (Drawing 40) are similarly sealed. Thermocouple leads pierce the fuel shield plugs and extend to the orifice area to measure the exiting sodium coolant temperature. In addition, each plug contains a small tube from which samples of reactor cover gas contained in the orifice tube can be monitored for fission products. By this means, a ruptured fuel element can be detected and located.

A single axial sodium inlet line supplies the coolant to the lower calandria plenum, where it is uniformly distributed to the process channels by a distribution nozzle. The sodium passes up through the core and exits via the orifice into the upper plenum. From the upper plenum it flows into a single outlet line through a perforated box header which eliminates vortexing and creates more uniform radial flow paths from the process channels.

A portion of the sodium is also supplied to the annulus between the reactor and calandria vessels for cooling of the vessel walls. The annular coolant flow is controlled by a valve which is automatically regulated to maintain uniform temperatures between the upper plenum sodium and the outlet coolant from the annulus. The annulus terminates in standpipes which protrude just above the sodium surface in the upper plenum in order to separate the annulus from possible variations in the free surface level that could cause adverse coolant circulation paths.

6. Fuel Element (Figure 3 and Drawing 41)

The nuclear fuel materials is hypostoichiometric uranium carbide enriched to 3.80 wt % U^{235} . The selection of uranium carbide as the fissile material is based on its compatibility with sodium, thermal properties which closely match the potential of sodium as a coolant, and satisfactory irradiation stability.

The fuel element consists of a bundle of 18 fuel rods and a center tube to support the fuel rods and fuel rod spacers. The fuel rods are 17 ft, 4 in. in length and are made up of 14 ft of uranium carbide slugs assembled in stainless steel tubes. The fuel slugs are 1/2 in. in diameter and vary in length from 1 to 8 in. The slugs are bonded to the tubes by sodium to improve heat transfer. The sodium annulus is 0.025 in. thick and allows for 10% local diametral swelling; the maximum expected local increase is only 8%. The design criterion for total volume increase is 17%, which is based on average diameter and length increases of 5.3%. According to these design criteria a 40-in.-long gas space is provided above the fuel within each tube to allow for expansion and fission gas collection. The design is such that the fission gas pressure at the operating temperature will not cause an accumulated cladding strain in excess of 1/2% throughout the life of the element.

The elements are designed to fit inside the 3.950-in.-ID process tubes and are supported at their lower end by the calandria bottom plate. Although the

Figure 3. Fuel Element

elements are heavy enough to offset the hydraulic uplift of the coolant, for additional safety they are latched into a seat at the bottom of the process channel. All structural material is 304 stainless steel. Design data for the fuel elements are listed in Table 3.

7. Control Element (Drawing 42)

The control elements serve as combined shim and safety rods. Incorporated in their design are the necessary features to provide adequate absorption material, vernier-type rod motion control, and quick insertion of poison by

TABLE 3
FUEL ELEMENT DESIGN DATA

Number of rods	18
Number of spacers	1 per ft of length
<u>Materials</u>	
Fuel	Hypo UC 4.5 - 4.8 wt % C
Clad	304 SS
Bond	Na
Enrichment (wt % U ²³⁵)	3.80
Uranium inventory (kg U/element)	124
<u>Fuel Rod</u>	
Fuel slug diameter (in.)	0.500 ± 0.005
Active fuel length (ft)	14
Fuel weight (kg U)	6.89
Sodium annulus (in.)	0.025
Clad thickness (in.)	0.010
Fuel rod diameter (in.)	0.570
Total fuel rod length (ft)	18 (nominal)
<u>Fuel Rod Design Criteria</u>	
Clad strain	0.5%
Fission gas release	3%
Na outlet temperature, maximum (°F)	1175
Fuel temperature, maximum (°F)	1750
Burnup average MWD/Mtu	25,000
Total ΔV	17%
Average Δd	5.3%
Average ΔL	5.3%

release and free fall with snubber action. There are a total of 24 control elements in the core which have a reactivity worth of 15% $\Delta k/k$. The elements provide a shutdown margin of ~ 5% $\Delta k/k$ for the proposed fuel loading. The absorber is designed for 10-yr operation, based on 85% capacity factor.

The neutron absorber material is powdered dysprosium oxide (Dy_2O_3) compacted to a density of 5.5 gm/cc. It forms a 3/16-in.-thick annular ring between an inner 2.125-in.-diameter solid cylinder of graphite and an outer Hastelloy-X tube which serves as the structural member. This composite structure moves vertically within a 3-in.-OD by 0.035-in. wall, gas-tight helium-filled thimble positioned in selected process channels. Thus, the absorber element does not contact the sodium coolant. An initial cold radial helium gap of 0.035-in. is provided between the absorber and thimble which permits free movement of the absorber during reactor operation, and allows adequate cooling of the absorber.

The control rod assembly is similar to that used in the Hallam Nuclear Power Facility. The absorber is lifted by three pull rods which pass through the loading face shield to the actuator disconnect section at the operating floor level. The control rods are moved by electric motor-driven actuators located on top of the loading face shield. The actuator drive motor and gearing is located at the very top of the rod assembly and rotates the ball screw shaft which lifts the ball nut and scram latch. The actuators are mounted so that they can be readily removed for fuel handling operations. The absorber section remains in the reactor during refueling. The absorber section is 13.0 ft in length and has a 13.0-ft travel within the active core. Normal speed of travel is 15-in./min.

Control elements are cooled by conduction of heat across the helium gap to the thimble wall, which is cooled by sodium flowing up through the annulus between the thimble and process tube. Since sodium coolant is provided from the high-pressure lower plenum, an orifice is placed at the bottom entrance of those channels containing control rods to reduce sodium flow to the desired rate.

Steady-state control of the reactor is provided by an automatic control system capable of moving several control rods simultaneously. The control system is such that all rods can be manually or automatically adjusted separately or in unison. (See Section VII for details of the plant control system.)

8. Miscellaneous Core Elements

The miscellaneous core elements include a startup neutron source, dummy elements, sodium level and temperature indicators, and neutron instrumentation elements.

9. Fuel-Handling System

In the fuel-handling procedure, an irradiated fuel element is taken directly from the active core into a lead-shielded cask which is an integral part of the fuel handling machine. The machine then transports the element either to another process tube or to the spent fuel storage pool. This is the same design philosophy that has been successfully demonstrated in SRE and Hallam (see Figure 4). Newly fabricated fuel is transferred to the fuel handling machine from a pickup cell. A schematic diagram of the fuel handling facilities is shown in Drawing 43.

a. Fuel-Handling Machine (Drawing 44)

The fuel-handling machine consists of a lead-shielded cask 60 ft high which incorporates fuel element and shield plug handling grapples and other auxiliary handling components, a fuel element cooling system, and an integral control console. The machine is mounted on a gantry (Drawing 45) spanning the high bay area of the reactor building, and travels on rails embedded in the floor. The gantry and all cask operations are controlled from the control console which is also mounted on the gantry.

The cask (Drawing 46) contains three cylindrical storage barrels, two for handling fuel elements and one for handling shield plugs or a control rod and thimble. The fuel-handling grapples can be extended approximately 25 ft below the loading face shield where they grapple a fuel element and actuate the latch at the base of the element. The shield plug grapple reaches only to the reactor loading face where it fastens to a shield plug or control rod thimble. Other details of the fuel handling cask are given in Drawing 47.

The fuel-handling barrels each contain a 4-in.-diameter guide tube which is lowered through the loading face shield with the grapple to the top of the process channel during the fuel transfer operation. It is held down in this position by two air-operated clamp devices located below the cask lower plenum. The fuel element is drawn into the guide tube and the guide tube is then drawn into the fuel handling cask by the grapple.

The fuel-handling machine has a helium cooling system which is capable of removing 25 kw from each of the fuel elements contained in the cask. The fuel element is cooled while in the guide tube by passing cool helium down

Figure 4. Hallam Fuel-Handling Machine

the annulus between the orifice tube and the guide tube. On the downmost position of the guide tube, the helium passes into the guide tube through holes just above the sodium level and then passes around the fuel rods and up the inside of the guide tube to a manifold outlet. It is then piped to a sodium vapor separator, heat exchanger, and helium blower for return to the cask. The blower system operates whenever an irradiated fuel element is loaded, discharged, or in the cask. Heat is transferred from the helium to the reactor room atmosphere by use of a 5-hp air blower. Two such systems are required for normal operation, and a third system is provided for standby purposes.

Three independent drives are used to raise or lower the grapples. Two of the units are identical for fuel-element handling, while the third has a larger capacity to accommodate the shield plug and control rod. Two motors are provided for each drive unit. The second motor in each case is used for standby purposes only.

The operation of all three drive units is identical. The grapple is supported and actuated by two tapes. These tapes travel over an idler wheel assembly to the driver pulleys. The pulleys are driven by a brake motor through an integral eddy current coupling.

The idler wheel assembly consists of two free-spinning pulleys mounted on an eccentric shaft. A 180° rotation of the shaft displaces the pulleys vertically. This action displaces the tapes, raising one slightly and lowering the other, thereby actuating the grapple. The idler assembly is supported by two load cells which register the load on each tape. An upper and lower limit on the load cell automatically stops the drive system.

b. Wet Fuel Storage (Figure 5 and Drawing 48)

Spent fuel is stored in liquid sodium so that the decay heat can be removed. Storage is provided by open-end tubes assembled in a turntable immersed in a sodium pool located below the operating floor about 20 ft from the reactor. The sodium pool is contained in a thermally insulated steel double-tank assembly ~ 12 ft in diameter and 30 ft high, which is suspended from the floor shielding structure into a cavity filled with nitrogen. A helium cover gas is maintained over the sodium pool. Details on the storage vessels are shown in Drawings 49, 50, 51, and 52.

Figure 5. Fuel Storage Vessel

The turntable (Drawing 53) accommodates 270 fuel elements and is supported from a large thrust bearing near the reactor floor level. The storage tubes are located and spaced so that the handling heads will be readily accessible through ports over the storage tank. The fuel-handling indexing and gripping requirements are similar for both this vessel and the reactor vessel. The element spacing is adequate to prevent criticality. Further details on the turntable are given in Drawing 54.

The storage vessel is capped by a 6-ft-thick concrete top shield (Drawing 55) which is penetrated by 6 ports, each capped by a shield plug (Drawing 56). Details of the top shield are shown in Drawings 57 and 58. The shield plugs provide access to six circumferential rows of storage tubes in the turntable. Access to any position is obtained by rotating the turntable until the desired storage tube is directly under one of the six shield plugs. An indexing mechanism allows selection and proper alignment for any given position.

Cooling of the fuel is accomplished by conduction and natural convection of the sodium within the vessel. The sodium pool is cooled by a separate system of NaK which flows by natural convection up through the 1-in. annulus between the walls of the double-tank assembly and down through an external loop to the bottom of the vessel. The NaK is cooled by a 1.2-Mw air heat exchanger which consists of a 1200-ft coil of finned tubing having 0.569 ft² of surface area/ft of length. The inlet and outlet piping of the NaK lines is heated with independent resistance heaters so that the temperatures can be adjusted to control the flow. By this means, the temperature of the sodium in the vessel is maintained in the range of 350 to 450°F.

c. New Fuel Storage

New fuel elements are stored in an air atmosphere in tubes in a storage room located below the operating floor level. The elements are lowered into place from the operating floor. Accommodation for 75 new elements is provided.

d. Control Rod Storage

The control rod storage facility consists of thirty-two 45.5-ft-long thimble sleeves suspended into a shielded cavity below the reactor floor and which are provided with shielded plugs and cover plates. The facility arrangement is shown in Drawing 43; and the thimble sleeve and plug designs, in Drawing 59. Control rods and their thimbles, removed from the reactor with the fuel

handling machine, are stored in these sleeves under a helium atmosphere. Cooling of control rods in storage is not required.

In addition to control rod storage, the cavity contains three special sleeves for miscellaneous reactor maintenance tools and reactor source element storage.

B. CORE PERFORMANCE

Major performance parameters for the reactor core and fuel elements are listed in Tables 4 and 5 respectively. The reactor core is designed to produce 480 Mwt with a maximum fuel centerline temperature of 1750°F and mixed-mean sodium inlet and outlet temperatures of 750°F and 1150°F, respectively. Approximately 94% of the heat is generated in the fuel and 6% outside the fuel, primarily in the moderator. However, essentially all of the energy generated in the core is removed by the fuel channel sodium coolant. The average specific power in the fuel is 23.0 kw/kg U.

TABLE 4
200-Mwe CORE PERFORMANCE DATA

Thermal power (Mwt)	480
Heat generated outside fuel (% of core power)	6
Heat removed by fuel channel coolant (% of core power)	100
Average specific power (kw/kg U)	23.0
Overall maximum to average power ratio	2.25
Total core coolant flow (lb/hr)	13.55 x 10 ⁶
Core pressure drop (psi)	21
Mixed-mean core outlet coolant temperature (°F)	1150
Inlet coolant temperature (°F)	750
Reactivity control requirements; hot, clean core (% $\Delta k/k$)	
Burnup (7.5 mo, 80% p. f.)	5.6
Equilibrium xenon	2.3
Equilibrium samarium group of fission products	0.9
Xenon transient override	0.6
Loss of sodium	3.5
	<hr/>
Total	12.9
Reactivity worth of control elements (% $\Delta k/k$)	15

TABLE 5

200-Mwe FUEL ELEMENT PERFORMANCE DATA

Initial enrichment (wt % U ²³⁵)	3.80
Final enrichment (wt % U ²³⁵)	1.60
Net plutonium production (g Pu/kg U)	6.86
Average fuel burnup (Mwd/MTU)	25,000
Average incore residence time - 80% p.f. (yr)	3.8
Maximum element power (Mwt)	4.3
Maximum to average element power ratio	1.49
Maximum linear rod power (kw/ft)	24.0
Maximum rod specific power (kw/kg U)	51.8
Maximum nominal heat flux at clad OD (Btu/hr-ft ²)	548,800
Maximum hot channel clad temperature (°F)	1250
Maximum hot channel fuel temperature (°F)	1750
Maximum element flow rate (lb/sec)	33.8
Maximum coolant velocity (ft/sec)	13.1

The initial enrichment required for the equilibrium core is 3.80 wt % U²³⁵. This requirement is based on the reactivity needs of the core to attain an average fuel element discharge burnup of 25,000 Mwd/MTU, to allow for temperature reactivity differences between operating and shutdown status, and to provide sufficient reactivity to override neutron poisons generated in the core, primarily xenon and samarium. At the end of design burnup, the final enrichment is 1.60 wt % and the net plutonium production is 6.86 g Pu/kg U.

Based on a fresh equilibrium core, the total reactivity control requirements are 12.9% $\Delta k/k$. A component breakdown of this total is given in Table IV-3. The 24 control elements provided in the core give a total reactivity control capability of approximately 15% $\Delta k/k$.

The fuel elements operate at a core mixed-mean coolant outlet temperature of 1150°F and a maximum inside clad surface temperature of 1250°F. Axial temperature profiles in the fuel, cladding, and sodium in the hot channel of the maximum powered fuel element are shown in Figure 6.

Figure 6. Axial Temperature Profiles in Fuel, Clad, and Sodium of Hot Channel in Maximum Power Fuel Element

The core pressure drop is determined by the required flow rate in the maximum power-producing element. This flow rate is 33.8 lb/sec and results in a maximum core coolant velocity of 13.1 ft/sec. At this velocity the sum of the frictional pressure loss, entrance and exit losses, and pressure loss through the open variable orifice results in a total core pressure drop of 21 psi.

Since core power is produced primarily from thermal neutron fissions, the reactor exhibits the proven easily controlled nuclear kinetics associated with thermal U^{235} systems. The inherent dynamic response characteristics of the core are governed by the prompt power coefficient which is determined primarily by Doppler-broadening of fuel resonance capture cross sections. This coefficient is negative and acts to compensate reactivity disturbances and reduce the magnitude of power changes from equilibrium conditions. Although there is a positive feedback from the coolant temperature, it is considerably smaller than the negative fuel effect. In addition, the short heat transfer time constants between fuel and coolant ensure that noticeable power oscillations after a power change do not occur.

Long-term or steady-state power coefficients of reactivity are positive over a portion of the operating range. This condition arises from the positive reactivity contribution of the moderator temperature, which follows (in time) the prompt coefficient effect after a power transient with a time constant on the order of 2 to 3 min. As a result of this long time constant relative to control system response, regulation of power level by the automatic control system is readily accomplished.

C. FUEL MANAGEMENT

1. Management Program

The fuel management program is designed to require partial replacement of fuel elements at the end of each 6.0 months of full-power operation at a 100% plant factor. In actual operation, the refueling interval is increased to match the actual time required to produce this quantity of energy. For example, for an average plant factor of 95%, a realistic value during the early years of plant life, the refueling interval is 6.3 months; for an 80% plant factor it is 7.5 months.

During each refueling, 28 fuel elements (1/6 of the core) are replaced with new elements, and 10 elements are relocated within the core to maintain a balanced power distribution. This relocation is from perimeter core positions to center core positions.

The fuel cycle is outlined in Table 6 for the first 5 yr of operation based on an 80% plant factor. The initial core loading consists of six batches of fuel at three different enrichment levels as shown. During each refueling, an amount of fuel equal to one batch is replaced. Following this program, the refueling schedule is the same during first core operation as it is during equilibrium core operation. The initial batches of fuel do not attain the full 25,000 Mwd/MTU, but are removed at lower burnup levels as specified in Table 6 in order to reach the equilibrium core cycle. For this reason, these batches require less initial enrichment and the fuel material will be correspondingly less expensive. On this basis, the equilibrium fuel cycle is reached after approximately 3-1/2 years operation. The average core residence time of the fuel during the equilibrium cycle is 3.8 yr at an 80% plant factor.

2. Refueling

a. Time Required

A 10-hr period is required after the reactor is shut down before fuel in the reactor can be handled. During this period all preparatory steps can be accomplished so that fuel movement can be initiated promptly. These preparatory steps include the following.

- 1) Reactor sodium temperatures are reduced to 650°F isothermal or lower.
- 2) Control rod drives are disconnected from their rods and the control rod drive carriage assembly is rolled to the side of the reactor. (The poison section of the control rods remains fully inserted in the core during refueling.)
- 3) The reactor floor is cleared, and the small cover plates over the fuel to be moved are unbolted and removed.
- 4) The thermocouple leads at the top of the fuel shield plugs are disconnected and moved to the side of the shield plug heads.
- 5) The fuel handling machine is started.

TABLE 6
FUEL CYCLE DATA FOR FIRST 5 YR OF OPERATION BASED ON
80% PLANT FACTOR

Batch Number	Approximate Time In/Out (months after startup)	Contained Fuel (kgU)	Average Discharge Exposure (Mwd/MTU)	Initial Enrichment (wt % U235)	Final Enrichment (wt % U235)	Final Plutonium Content (gm/kgU)
1*	0/8	3,472	4,600	2.20	1.72	2.33
2*	0/15.5		8,800	2.20	1.38	3.71
3*	0/23		12,500	3.00	1.81	4.66
4*	0/30.5		16,700	3.00	1.51	5.51
5*	0/38		20,400	3.80	1.92	6.26
6*	0/45.5		24,500		1.63	6.75
7	8/53		25,000		1.60	6.86
8	15.5/60.5		25,000			
9†	23/68		25,000			
10†	30.5/75.5		25,000			
11†	38/83		25,000			
12†	45.5/90.5		25,000			
13†	53/98		25,000			
14†	60.5/105.5		25,000			

*These batches comprise the initial core loading

†These batches would remain in the core beyond the 5-yr period

An additional preparatory step is accomplished during reactor operation approximately a week before shutdown. This step is the relocation of the new fuel elements which are to be inserted in the core, from the new fuel storage room to the wet fuel storage pool. This is done to reduce the fuel handling time per element during the refueling operation and to preheat the new elements before insertion in the hot reactor sodium.

At HNPF, a spent fuel element can be replaced with a new element from the storage area in less than 1.5 hr. The time required to relocate an element within the core of the 200-Mw plant is estimated to be 1 hr because of the smaller distance to be covered by the fuel-handling machine and the fewer steps required. Therefore, the normal refueling operation, requiring replacement of 28 elements and relocation of 10 elements, requires about 52 hr. Including the 10 hr cooling period and startup, the total shutdown time is approximately three days.

b. Fuel-Handling Operations

(1) Receiving New Fuel

New fuel rods are delivered by truck to the receiving ramp and moved into the building at ground level. The rods are assembled into fuel elements in an area adjacent to the reactor operating room. After assembly, the element is moved to the reactor operating room and positioned above the new fuel storage room. The floor cover to the new fuel storage room is removed and the new fuel element is lowered into tubes for storage until reactor refueling.

(2) Reactor Refueling

The turntable in the wet fuel storage vessel is rotated so that a new fuel element is in line with one of the six vessel shield plugs, and then is locked into position. An adapter ring is placed over the shield plug to be removed. The fuel-handling machine is moved and indexed over this shield plug. The fuel-handling cask seal valve is lowered onto the adapter ring and an O-ring provides a seal between them. The space between the cask seal valve and the top of the shield plug is purged and filled with helium. The movable radiation shield on the cask is lowered and the cask seal valve opened. The steps outlined in this paragraph are repeated whenever the fuel-handling machine is connected to a fuel-element location in either the reactor or wet fuel storage vessel. Therefore, these steps will not be repeated throughout the remainder of this procedure description.

The fuel-handling machine cask drive assembly has three pickup positions, one for shield plugs and two for fuel assemblies. The drive assembly is rotated to the shield plug position and the shield plug grapple is lowered and latched onto the shield plug. The shield plug is lifted into the cask. (A load cell indicates to the operator when a load is attached to the grapple.) On up-position of the grapple, the drive assembly is rotated 120° to a fuel pickup position. The fuel grapple and the coolant guide tube are lowered through the shield plug opening into the storage vessel and the grapple is latched onto the fuel assembly. The new fuel element is lifted through the guide tube and into the cask. Simultaneously, the guide tube is also lifted into the cask by the grapple. On the up-position, the drive assembly is again rotated 120° and locked into position. The shield plug is lowered back into position, the cask seal valve is closed, and the cask movable shield is raised.

The fuel-handling machine is now moved to the reactor loading face shield. Reactor temperature and sodium flow have been reduced for fuel transfer. Before proceeding further, the helium cooling system of the fuel handling machine is started and verified to be functioning properly. An irradiated element is never handled without this coolant system in operation. It is assumed at this point that the steps preparatory to refueling, as outlined in the previous section, have been completed.

An adapter pickup is attached to the fuel shield plug which is to be removed to permit handling with the machine grapple. An adapter ring is inserted into the annulus around the top of the shield plug. The fuel-handling cask valve seals to the flange on this adapter ring during fuel removal and insertion. The drive assembly is rotated to the shield plug position and the reactor shield plug is removed. The drive assembly is rotated to the empty fuel-handling position, the fuel grapple and guide tube are lowered, and the spent fuel element drawn into the cask. Helium cooling is provided for the element throughout this step. The drive assembly is then rotated to the barrel containing the new fuel element, the element is lowered into the reactor, and the grapple is unlatched and lifted back into the cask. The drive assembly is rotated to the plug position and the shield plug is lowered into place over the new fuel.

The turntable in the wet fuel storage vessel is rotated to an empty position below one of the six floor plugs, and locked into position. An adapter ring, similar to the one used over the reactor, is placed over the storage plug. The fuel-handling machine is moved and indexed over this storage plug, which is then lifted into the cask. The drive assembly is then rotated so that the spent fuel is directly over the storage hole. The spent fuel is lowered into the turntable in the storage tank, and the grapple is unlatched and raised into the cask. The storage plug is lowered back into place over the storage tank, completing the fuel element replacement.

(3) Shipping Spent Fuel

Spent fuel is shipped in a lead-shielded fuel shipping cask which is transported by truck directly into the reactor room. The overhead crane raises the shipping cask from the truck and places it into the cask receiving pit in the operating floor.

The fuel-handling machine is indexed over a spent fuel element in the wet fuel storage vessel and the shield plug is lifted into the fuel-handling machine cask. The drive assembly is rotated to a fuel-handling position, and the spent element is lifted into the cask. Cooling is not required at this time because the fuel has completed its decay period.

The fuel-handling machine is moved over the plug in the shipping cask. The plug is raised into the fuel-handling machine cask. The spent fuel in the cask is rotated over the plug hole and lowered into the shipping cask. The rack in the shipping cask is then rotated to an empty position to receive another element. This procedure is repeated until the shipping cask is filled (~ 7 elements). The shipping cask is then sealed and placed in the truck for transport.

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V. SODIUM HEAT TRANSFER SYSTEM

The sodium heat transfer system removes the heat generated in the reactor core and delivers it to steam generators for production of steam. Heat transfer is accomplished by circulating liquid sodium in a main and an auxiliary circuit. Two independent circuits are required to assure removal of reactor decay heat after a scram. Nominal thermal capacity of the entire system is 480 Mwt. The main circuit transfers 432 Mwt and the auxiliary circuit transfers 48 Mwt. A diagram of the sodium heat transfer system is shown on Drawing 60; flowrates and equipment relative elevations are shown in Drawing 61. Major design data for the system are tabulated in Table 7.

Each circuit consists of a radioactive sodium primary loop which transports thermal energy from the reactor to intermediate heat exchangers (IHX), and a nonradioactive sodium secondary loop which transports heat from the intermediate heat exchangers to the steam generator. The intermediate heat exchangers are utilized to separate the radioactive sodium of the primary loop from the steam generators. This is done to (a) permit access to the steam generating facilities during operation, and (b) protect the reactor from pressure and temperature surges which could result from a steam or water leak in the steam generator to the reactor coolant. Each circuit incorporates separate steam generator banks which consist of sodium heated evaporators, and sodium-to-steam superheaters. The main circuit is also provided with sodium-to-steam reheaters.

The capacity of the auxiliary circuit is sufficient for decay heat removal under forced convection conditions. In addition, the main intermediate heat exchangers and steam generator banks are arranged vertically, so that a natural convection head will exist in the main system to remove decay heat in the event of power loss or other malfunction.

Because the primary sodium is radioactive, all primary loop components are located within shielded equipment spaces, called cells, in the reactor building. All secondary nonradioactive system components, such as the steam generators and secondary sodium pump, are located outside and adjacent to the reactor building. The operating pressure on the secondary sodium side (shell side) of the intermediate heat exchangers is maintained higher than on the

TABLE 7

200-Mwe SODIUM HEAT TRANSFER SYSTEM

MAIN CIRCUIT

Thermal capacity (nominal Mwt)	432	
No. of loops	1 primary; 1 secondary	
No. of intermediate heat exchangers	2	
No. of steam generator banks	2	
Primary loop		
Sodium temperature (°F)		
Reactor inlet	750	
Reactor outlet	1150	
Sodium flowrate (lb/hr)	12.2×10^6	
Pump capacity (gpm)	30,700	
Total dynamic head (ft)	190	
NPSH available (ft)	28	
Motor power (hp)	2,000	
Pipe sizes (nominal in.)		
Pump suction	30	
Pump discharge	28	
Cold leg	26	
Secondary loop		
Sodium temperature (°F)		
Steam generator inlet	1075	
Steam generator outlet	675	
Sodium flowrate (lb/hr)	12.2×10^6	
Pump capacity (gpm)	30,700	
Total dynamic head (ft)	190	
Motor power (hp)	2,000	
Pipe sizes (nominal in.)		
Hot leg	28	
Cold leg	26	
Intermediate heat exchanger		
Number of units	2	
Type construction	Modular shell and tube unit	
Duty per unit (Btu/hr)	7.4×10^8	
Surface area per unit (sq ft)	10,900	
Tube data per unit	3510 tubes, 5/8 in. OD by 0.049 in. wall, 19 ft long	
	<u>Tube Side</u>	<u>Shell Side</u>
Fluid	Primary Na	Secondary Na
Flow per unit (lb/hr)	6.1×10^6	6.1×10^6
Temperature in (°F)	1150	675
Temperature out (°F)	750	1075
Design pressure (psig)	100	100
Design temperature (°F)	1200	1200
Pressure drop (psi)	~7	~4
Material	304 SS	304 SS

2

TABLE 7 (Continued)

AUXILIARY CIRCUIT

Thermal capacity (nominal Mwt)	48	
No. of loops	1 primary; 1 secondary	
No. of intermediate heat exchangers	1	
No. of steam generator banks	1	
Primary loop		
Sodium temperature (°F)		
Reactor inlet	750	
Reactor outlet	1150	
Sodium flowrate (lb/hr)	1.35×10^6	
Pump capacity (gpm)	3400	
Total dynamic head (ft)	190	
NPSH available (ft)	28	
Motor power (hp)	250	
Pipe sizes (nominal in.)		
Pump suction	12	
Pump discharge	10	
Cold leg	10	
Secondary loop		
Sodium temperature (°F)		
Steam generator inlet	1075	
Steam generator outlet	675	
Sodium flowrate (lb/hr)	1.35×10^6	
Pump capacity (gpm)	3400	
Total dynamic head (ft)	190	
Motor power (hp)	250	
Pipe sizes (nominal in.)		
Hot leg	10	
Cold leg	10	
Intermediate heat exchanger		
No. of units	1	
Type construction	Modular shell and tube unit	
Duty per unit (Btu/hr)	1.66×10^8	
Surface area per unit (sq ft)	2425	
Tube data per unit	781 tubes, 5/8 in. OD by 0.049 in. wall, 19 ft long	
	<u>Tube Side</u>	<u>Shell Side</u>
Fluid	Primary Na	Secondary Na
Flow per unit (lb/hr)	1.35×10^6	1.35×10^6
Temperature in (°F)	1150	675
Temperature out (°F)	750	1075
Design pressure (psig)	100	100
Design temperature (°F)	1200	1200
Pressure drop (psi)	~7	~4
Material	304 SS	304 SS

primary sodium side (tube side) by an inert gas atmosphere at the expansion tank so that in the event of a leak in one of the IHX tubes the unshielded secondary system will not become contaminated with radioactive sodium.

Sodium purity in the heat transfer system is controlled by hot traps and cold traps which remove carbon and sodium oxide, respectively, from the sodium. Oxide concentration is continuously monitored by plugging meters installed in the bypass sodium stream to the purification equipment. This equipment is described in greater detail in Section VIII.

A. MAIN HEAT TRANSFER CIRCUIT

The main primary loop takes suction from the pool of sodium above the core in the reactor vessel. The sodium is pumped through the intermediate heat exchanger and through the reactor by a 30,700-gpm centrifugal pump.

The primary sodium pump is a vertical free-surface, single unit, consisting of an upper barrel assembly which contains both shielding and thermal insulation, an expansion space for sodium to rise into the pump barrel, a hydraulic radial bearing, and the pump impeller and diffuser. The entire pump inner assembly is arranged so that it may be removed without disturbing the outer pump casing or piping connections. The upper radial and thrust bearings are conventional bearings, cooled and lubricated by the pump seal oil system. A double mechanical seal is used at the upper, cold end of the pump for atmospheric separation. The pump is driven by a 2000-hp motor connected to a variable speed drive for flow control. A typical primary sodium pump, of auxiliary circuit size, is shown in Figure 7. The main circuit primary sodium pump will be of similar configuration except that the motor and variable speed drive will be horizontally mounted because of the larger size.

The pump suction line of the main primary loop is 30-in. pipe and the discharge line is 28-in. The cold return line to the reactor vessel is 26-in. Piping material is Type 304 stainless steel.

The piping configuration to and from the reactor reduces to acceptable levels the neutron streaming into equipment spaces and the resultant activation of equipment. The suction piping is double-walled from the reactor vessel until it reaches a height equal to the liquid level inside the reactor so that in case of a major line rupture, exposure of the core is prevented. In addition, a high

Figure 7. Typical Auxiliary Circuit
Primary Sodium Pump

point is provided in the suction and return piping, which is higher than the reactor core and the reactor vessel outlet nozzle, to permit the main primary loop to be drained without draining the auxiliary primary loop. This allows decay heat removal during shutdown with the auxiliary circuit, and at the same time, access after a decay period to the main primary loop without removing fuel from the reactor. Double-walled piping is also provided around the return line, from the reactor vessel back to the first valve.

A check valve and butterfly valve are provided in the return line to the reactor. Because of the parallel operation of the pumps in the main and auxiliary primary loops, a check valve is needed in each primary loop to prevent flow reversal in the event of loss of a pump or mismatch of pump speeds. The butterfly valve is used for flow control following a scram in order to prevent overcooling of the reactor and rapid loss of the reactor temperature gradient.

The main and auxiliary primary loops are located in separate cells. The dividing wall between cells is primarily a structural wall, but it also affords some shielding for access to a drained loop prior to complete decay of a filled loop in the adjacent cell. A drained loop is accessible 10 days after shutdown, whereas for a filled loop an additional 3-1/2 days decay is necessary to reach the same radiation level.

Two intermediate heat exchangers (IHX) operate in parallel in the main circuit. The heat exchangers are vertical shell and tube modules approximately 60 in. in diameter and 27 ft long. Each unit has a surface area of 10,900 ft². An expansion joint in the shell allows for differential thermal expansion between the shell and tube bundle. Figure 8 shows a typical intermediate heat exchanger.

The main secondary loop has the same thermal capacity as the main primary loop. Secondary sodium is pumped in a closed loop through the shell side of the intermediate heat exchangers and steam generating facilities. Pumping capacity is identical with the primary loop. The secondary sodium pumps are the same type as the primary, but a radiation shield is not required. The piping in the hot-leg (IHX to steam generator) is 28-in. OD; the cold-leg line (steam generator to IHX) is 26-in. OD. An expansion tank floats on the line to absorb system pressure surge and volume changes in the sodium due to temperature variations.

Figure 8. Typical Intermediate Heat Exchanger

B. AUXILIARY HEAT TRANSFER CIRCUIT

The auxiliary heat transfer circuit transports approximately 10% of core output during full-power operation and provides for decay heat removal after reactor shutdown. The auxiliary primary loop takes suction from the main primary loop suction line at an elevation below the minimum liquid level of the reactor, and rises above the minimum liquid level. All components in the auxiliary primary loop are located above this minimum liquid level so that a loss of loop integrity cannot cause core exposure. Double-walled piping is provided on the suction line up to the high point in the line and on the return line back to the first valve.

The auxiliary primary sodium pump is rated at 3400 gpm with a 250-hp variable speed drive motor. The suction line is 12-in. OD pipe; and the discharge line is 10-in. OD.

Only one intermediate heat exchanger is provided in the auxiliary circuit. It is similar to and smaller than the one in the main circuit, with a surface area of 2425 ft². Additional details are given in Table 7.

The secondary loop of the auxiliary circuit has the same thermal and pumping capacity as the primary loop. Piping is 10-in. and the secondary sodium flows through the shell side of the IHX and steam generator. An expansion tank is again provided to allow for sodium volume and pressure changes.

C. STEAM GENERATOR

In the steam generator the secondary sodium transfers its heat to water and steam to produce superheated steam at turbine throttle conditions of 2400 psig and 1000°F. The steam generator is made up from three types of modular units: (1) low-temperature evaporator modules (Drawing 62), (2) high-temperature superheater modules (Drawing 63), and (3) reheater modules (Drawing 64). These are mounted vertically in a once-through design with sodium on the shell side and steam on the tube side. Secondary sodium enters the top of the high-temperature modules (superheater and reheater) and flows downward through the shell. The sodium leaving the reheater modules mixes with the sodium leaving the superheater module, and the total combined sodium flow enters the evaporator modules. The sodium exits from the evaporator modules and returns to the intermediate heat exchanger.

The module concept provides a reliable, low cost steam generator design by:

- 1) Dividing the steam generator into a high-temperature section which is fabricated from austenitic stainless steel and a low-temperature section which is fabricated from less expensive chrome-moly steel. This division also alleviates the chloride stress corrosion problems associated with austenitic steel.
- 2) Employing butt-weld attachment of tubes to the tube-sheet face. This eliminates crevices and their associated corrosion problems for austenitic steels.
- 3) Using commercial sizes of chromium-molybdenum and stainless steel pipe and tube fittings. This minimizes fabrication cost.

The steam generator modules are arranged in three banks, two banks in the main circuit and one in the auxiliary circuit. Each bank is arranged in a circular array, with superheater modules above the evaporator modules. The reheater modules, which exist in the main circuit only, are placed at the side of the superheater modules. The modules are mounted vertically in order to minimize horizontal space requirements in the plant and to avoid sodium stratification problems during free convection. The auxiliary circuit bank is located between the two main banks. The steam generator modules and the steam, water, and sodium heater systems are supported in a steel structure which is enclosed by sheet metal siding. Pressure relief headers run vertically from the modules to pressure relief tanks. A typical modular steam generator bank is shown in Figure 9; design information is listed in Table 8. Arrangement plans and sections are shown in Drawings 65 and 66.

Each steam generator bank in the main circuit consists of thirteen low-temperature evaporator modules, eight high-temperature superheater modules, and three reheater modules. The banks are interconnected on the sodium and steam sides, thus permitting parallel operation under any load conditions. The auxiliary circuit steam generator bank consists of three low-temperature evaporator modules and two high-temperature superheater modules. There is no reheater in the auxiliary circuit.

The evaporator and superheater modules are similar in size and construction, being nominally 8 in. in diameter and utilizing 61 tubes. The reheater

- 1 HIGH TEMP STEAM OUTLET
- 2 REHEAT STEAM INLET
- 3 REHEAT STEAM OUTLET
- 4 FEED WATER INLET
- 5 MAIN SODIUM INLET
- 6 AUXILIARY BANK HIGH TEMP MODULES
- 7 AUXILIARY BANK LOW TEMP MODULES
- 8 MAIN BANK HIGH TEMP MODULES
- 9 MAIN BANK LOW TEMP MODULES
- 10 MAIN BANK REHEAT MODULES
- 11 AUXILIARY PRESSURE RELIEF TANK
- 12 MAIN PRESSURE RELIEF TANK
- 13 MAIN SODIUM OUTLET
- 14 DUMP STACK
- 15 AUXILIARY SODIUM INLET
- 16 AUXILIARY SODIUM OUTLET

Figure 9. Modular Steam Generator

TABLE 8
MODULAR STEAM GENERATOR

	Main Circuit		Auxiliary Circuit	
<u>Evaporator (low-temperature module)</u>				
Type of construction	Shell and tube		Shell and tube	
No. of banks	2		1	
No. of modules per bank	13		3	
Duty per module (Btu/hr)	31.5 x 10 ⁶		36.5 x 10 ⁶	
Surface area per module (sq ft)	476		433	
Tube data per module	61 tubes, 5/8-in. OD 0.095-in. wall, 47.6 ft long		61 tubes, 5/8-in. OD 0.095-in. wall, 43.3 ft long	
	<u>Tube</u>	<u>Shell</u>	<u>Tube</u>	<u>Shell</u>
Fluid	Water/steam	Secondary Na	Water/steam	Secondary Na
Flow per module (lb/hr)	45,800	459,600	53,200	443,000
Temperature in (°F)	473	900	473	945
Temperature out (°F)	695	675	695	675
Inlet feedwater pressure (psig)	2700	-	2700	-
Design pressure (psia)	2800	100	2800	100
Design temperature (°F)	975	975	975	975
Material	5 Cr-1/2 Mo-T1	5 Cr-1/2 Mo-T1	5 Cr-1/2 Mo-T1	5 Cr-1/2 Mo-T1
<u>Superheater (high-temperature module)</u>				
Type of construction	Shell and tube		Shell and tube	
No. of banks	2		1	
No. of modules per bank	8		2	
Duty per module (Btu/hr)	23.7 x 10 ⁶		25.5 x 10 ⁶	
Surface area per module (sq ft)	550		484	
Tube data per module	61 tubes, 5/8-in. OD 0.095-in. wall, 55.0 ft long		61 tubes, 5/8-in. OD 0.095-in. wall, 48.4 ft long	
	<u>Tube</u>	<u>Shell</u>	<u>Tube</u>	<u>Shell</u>
Fluid	Steam	Secondary Na	Steam	Secondary Na
Flow per module (lb/hr)	74,500	453,100	79,700	664,000
Temperature in (°F)	695	1075	695	1075
Temperature out (°F)	1005	900	1005	945
Outlet steam pressure (psia)	2500	-	2500	-
Design pressure (psia)	2800	100	2800	100
Design temperature (°F)	1100	1100	1100	1100
Material	321 SS	321 SS	321 SS	321 SS
<u>Reheater</u>				
Type of construction	Shell and tube		None required	
No. of banks	2			
No. of modules per bank	3			
Duty per module (Btu/hr)	41.1 x 10 ⁶			
Surface area per module (sq ft)	2030			
Tube data per module	187 tubes, 1-in. OD 0.049-in. wall, 41.5-ft long			
	<u>Tube</u>	<u>Shell</u>		
Fluid	Steam	Secondary Na		
Flow per module (lb/hr)	197,300	783,300		
Temperature in (°F)	630	1075		
Temperature out (°F)	1005	900		
Outlet steam pressure (psia)	500	-		
Design pressure (psia)	600	100		
Design temperature (°F)	1100	1100		
Material	321 SS	321 SS		

module is similar in construction but is larger, with a nominal diameter of 29 in. and 187 tubes.

The low-temperature module tubing is standard drawn, 5% chrome-1/2% moly-titanium stabilized, and the high-temperature module tubing is Type 321 stainless steel. The evaporator and superheater tubing is 5/8-in. OD with a 0.095-in. wall thickness; reheater tubing is 1-in. OD with an 0.049-in. wall.

Bosses are bored in the tube sheets to form attachments for the tubes. The tubes are then welded to the tube sheet using "inside out" techniques now well developed by several manufacturers. This type of tube to tube sheet attachment eliminates the crevice formed in a rolled and welded joint. End bells for feedwater and steam are formed by welding forgings to the tube-sheet forgings. The tube bundle is enclosed by a shell of standard piping with sodium inlet and outlet nozzles formed by standard piping tees. To minimize thermal shock during sodium temperature transients, an inert-gas blanket is maintained at the top of the module and a series of thermal baffles is provided at the bottom.

Each evaporator, superheater, and reheater module has a hydrogen detector mounted in the sodium outlet piping to detect possible water leakage into sodium. In the unlikely event of a leak, it will normally be detected and the affected bank shutdown before the leak becomes significant. However, in the event of a large leak, rupture discs are provided to vent the effluent products. The gaseous products are vented from the blowdown stack and the sodium is collected in pressure relief tanks.

VI. TURBINE-GENERATOR PLANT

An important feature of the turbine-generator plant is the use of a modern reheat cycle with steam conditions of 2400 psig/1000°F/1000°F. The net station heat rate of 8160 Btu/kwh, as shown on the heat balance diagram (Drawing 67), represents a station efficiency of 41.8%. The SGR system, with its high thermal performance, is capable of utilizing modern steam conditions to great advantage, resulting in the highest station efficiency of any nuclear or fossil-fuel-fired power plant currently in operation or under construction. The turbine-generator plant, described in this section, employs conventional type equipment of standard designs that have proven reliability based on years of operating experience in the power industry.

A. TURBINE-GENERATOR UNIT

The estimated net turbine cycle heat rate as a function of load for the 200,000-kw tandem compound, double-flow, 30-in. last-stage bucket machine selected for the 200-Mwe prototype plant is shown in Figure 10.

Figure 10. Cycle Heat Rate as Function of Load

The main steam throttle conditions will be 2400 psig at 1000°F and reheat to 1000°F, with a rated capability of 210,049 kw at 1-1/2 in. Hg abs back-pressure with 0% makeup and extraction for 7 stages of feedwater heating. The ultimate expected capability is approximately 221,000 kw with throttle valves wide open and the same steam and exhaust conditions. Controls will be designed for constant pressure operation at the inlet to the turbine throttle at 2400 psig.

The generator rating is 256,000 kva at 3600 rpm and 45 psig H₂ pressure with conductor-cooled armature and field windings. Additional turbine generator performance data are presented in Table 9.

TABLE 9
TURBINE- GENERATOR DATA

Rated generator gross electrical output (kw)	210,049
Ultimate generator gross electrical output (kw)	221,000
Rated net turbine cycle heat rate (Btu/kwh)	7830
Ultimate net turbine cycle heat rate (Btu/kwh)	7844
Throttle steam flow at rated capability (lb/hr)	1,351,700
Reheat steam flow at rated capability (lb/hr)	1,183,800
Condenser steam flow at rated capability (lb/hr)	923,560
Main steam pressure (psig)	2400
Reheat steam pressure (rated psig)	484.8
Initial throttle and reheat steam temperature (°F)	1000
Turbine exhaust pressure (in. Hg abs)	1-1/2
No. of stages of feedwater heating	7
Final feedwater temperature (rated- °F)	472.5
Generator kva rating at 45 psig H ₂	256,000
Generator preferred terminal voltage (kv)	22.0
Power factor	0.85
Short-circuit ratio	0.64
Preferred field voltage (v)	375

1. Turbine Design Features

The turbine consists of a high-pressure machine, an intermediate pressure machine, and a dual exhaust low-pressure machine mounted on a single shaft and operating at 3600 rpm. Superheated steam at 2500 psia/1005°F from the high-temperature steam generator modules enters the high-pressure turbine at 2400 psig/1000°F and exhausts as cold reheat steam at 555 psig/630°F. The reheater steam generator modules raise the cold reheat steam temperature to 1005°F, so that the hot reheat steam entering the intermediate turbine is at 499.5 psia/1000°F. Steam from the intermediate turbine exhausts at 85.8 psia and crosses over to the dual-flow low-pressure turbine which exhausts to the main steam condenser.

The turbine shaft penetration through each turbine shell element is sealed by steam against leakage to atmosphere by means of labyrinth-type seals. Steam which leaks past the shaft labyrinth penetrations is directed into a gland steam header which provides steam for vacuum seals.

2. Generator Design Features

The generator is a 3600-rpm synchronous unit rated at 256,000 kva, 0.85 power factor, 210,049 kw at 45-psig hydrogen pressure. The generator is built with a liquid-cooled armature winding. Hydrogen coolers remove heat from the hydrogen gas which is circulated through the armature core and the rotor by conventional axial flow fans. The liquid coolant is circulated through the windings by means of external equipment, consisting of the liquid cooler heat exchanger, filter, reservoir, and circulating pumps. Condensate from the main condenser is the cooling medium for the hydrogen coolers and the liquid cooler heat exchangers.

3. Auxiliaries and Accessories

The normal complement of accessory and auxiliary equipment including such items as supervisory instruments, exciter, lube oil system, pumps and valves are furnished. In addition, two cold reheat stop valves are required besides the conventional steam flow control valves in order to isolate the turbine from the reheater.

B. CONDENSING SYSTEM

The condenser will be a rigidly supported single-pass divided water box unit, with 68,500 ft² of tube surface. Approximately 87,600 gpm of circulating water will flow through the 7/8-in.-OD, 18-BWG tubes to condense 9.24×10^5 lb/hr steam and produce a back pressure of 1.5 in. of Hg abs with a cooling water inlet temperature of 57°F. The tube cleanliness factor is 85%; water velocity is 7.5 fps.

The condenser hotwell is compartmented into 4 sections for ease of tube leakage monitoring and provides for 5-min operation at rated load. The system is complete with steam-jet ejectors for hogging and normal operation, and two half-capacity condensate pumps. Each pump is rated at 1550 gpm and a TDH of 820 ft and is of vertical, multistage, motor-driven centrifugal design.

C. CIRCULATING WATER SYSTEM

A reinforced concrete intake structure is located at the river immediately west of the turbine building. The intake structure contains the circulating water pumps, traveling water screens, bar racks, service water pumps, screen wash pumps, and fire pump. The structure is open-air except for a heated shelter over the auxiliary pumps and the diesel engine driver for the fire pump. Shelters are not provided for the circulating water pumps. Bar racks are galvanized carbon steel and are cleaned with a movable mechanical trash rake. The rake runs on stainless clad carbon steel rails. Wooden stop logs are provided for installation in slots in the concrete walls. All equipment may be serviced by a mobile yard crane. De-icing this structure is achieved by partial recirculation of discharge water.

The trash rake consists of a hoisting structure with an integral apron, a 4000-lb-capacity double-trim single motor hoist and separate trash car. The traveling screens are single-speed outdoor type. The screens are set to operate simultaneously and the screen wash cycle is initiated by a differential controller for each screen, or by one preset time clock. The control panel is located in the chlorine house.

An 8000-lb-per-day chlorinator is installed in the chlorination building adjacent to the intake structure. A control system enables the chlorinator to treat the circulating and service water to a predetermined dosage by injecting the chlorine in the intake structure forebay. Chlorine is supplied to the chlorinator from a chlorine tank car located at the chlorine unloading station.

Two vertical, wet-pit, motor-driven, centrifugal pumps supply water for screen washing. Each pump is rated at 625 gpm at a TDH of 250 ft, and has capacity sufficient to wash both screens simultaneously. A cross-connection is provided between the screen wash system and the plant fire system so that the screen wash pumps can provide emergency backup to the diesel engine fire pump.

Two half-capacity, vertical, mixed flow, motor-driven, wet-pit type pumps with removable elements and weather protected motors supply circulating water from the river to the main condenser. Each pump is rated at 43,800 gpm at a TDH of 25 ft.

Two 48-in. ID steel inlet lines convey water from the circulating water pumps to the condenser. An expansion joint and motorized butterfly valve are provided in each line. Each line between the intake structure and turbine building wall is encased in 12 in. of lean concrete.

A small mechanical vacuum pump is provided for scavenging air and gases from the top of the water boxes during operation.

Circulating water from the condenser to the turbine building wall is carried in two 48-in. steel lines which join to form a single 68-in. ID steel line. A 68-in. reinforced concrete pipe conveys the discharge water from the building wall to the seal well. Reinforced concrete anchor blocks are used at the pipe-ends and at all elbows in both the intake and discharge lines.

A seal well is used to maintain a water seal at the end of the discharge line. It is located on the river bank and discharges directly into the river sufficiently downstream from the intake to prevent water recirculation. The seal-well is a reinforced concrete structure.

Expansion joints are provided in each of the water box connections.

D. CONDENSATE SYSTEM

The steam and condensate flow diagram shown in the heat balance Drawing 67 indicates the basic arrangement. Two half-capacity condensate pumps take suction from the condenser hotwell and discharge through the condensate demineralizer, air ejector inter- and after-condensers, gland steam condenser, and three low-pressure heaters into the main deaerator.

Excess condensate is returned to the storage tank by the condensate pumps through hotwell high-level control. A makeup line from the tank is provided to supplement the condensate supply at low water-level in the hotwell. The main condenser condensate circulation control is actuated by the differential pressure across the inter- and after-condensers to recirculate condensate at low loads in order to provide the minimum amount of water required for the air ejector condensers and the gland steam condenser.

Three low-pressure feedwater heaters are provided. All heaters are horizontal-mounted U-tube type, with removable tube bundles. The two lowest pressure heaters are mounted in the condenser neck. All of the heaters contain integral drain coolers with the exception of Heater 7, which discharges into the main feedwater stream with a drain pump. Drains from the three L.P. heaters are pumped into the main feedwater stream between L.P. Heaters 6 and 7, with Heater 7 drain pump. Each heater is provided with a shell-side vent line to the steam space of the condenser. Each heater drain line contains a control valve actuated by level controllers on the heaters to maintain the proper liquid level in the feedwater heater.

A demineralizer is provided in the condensate system immediately downstream of the condensate pumps to maintain the purity of the system. The demineralizer consists of three 1000-gpm-capacity mixed bed units, one cation regeneration unit, one anion regeneration unit, and a resin storage vessel.

Hydrazine and amine are injected into the condensate system downstream of the condensate demineralizer for scavenging free oxygen and controlling pH.

One 100,000-gal cone roofed storage tank of aluminum construction is provided for storage of excess condensate, demineralized makeup water, and recovered water from the waste treatment system.

E. FEEDWATER SYSTEM

Two half-capacity boiler feedwater pumps, driven by electrical motors through fluid coupling are provided to pump water through the three high pressure feedwater heaters to the steam generator. The boiler feedwater pumps are barrel type, centrifugal units rated at 1550 gpm and a TDH of 7500 ft. Lubrication for the main boiler feedwater pumps is supplied from their fluid couplings. Both pumps are furnished with minimum flow recirculation lines which

discharge to the main condenser through restricting orifices. Control valves in the recirculation lines are actuated by a flow switch which receives its signal from flow orifices in the pump suction lines.

Three high-pressure feedwater heaters are provided. They are horizontal-mounted, U-tube type with removable tube bundles. The heaters contain desuperheating and drain cooling sections; the tubes are carbon steel. Connections are provided for admitting nitrogen to blanket both the channel and shell sides during outages.

Drains from the 3-hp Heaters 1, 2, and 3 cascade to the deaerating Heater 4.

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VII. CONTROL AND INSTRUMENTATION

Station control is designed to permit automatic load-following operation over the range of 20 to 110% of rated load, with semiautomatic operation of the plant at any load down to no-load, hot-standby condition. Startup and shutdown are semiautomatic (combination of manual and automatic control).

The central control room is located at the turbine operating floor level. Master station controls are located here and comprise controls for the turbine-generator and switchyard, controls for heat generation and transfer, and some special panels (of a general rather than local nature). Examples of these panels are protection, radiation, fire, preheat, and reactor scanner. Local control panels are provided outside the central control room for certain process systems.

The discussion in the following sections covers control of the reactor and heat-transfer systems (plant control system) and certain special instrument systems important or unique to a nuclear facility (nuclear, protective, and radiation systems).

A. PLANT CONTROL SYSTEM (Drawing 68)

The plant control system is comprised of two main subsystems: one for control of the reactor and the other for main process heat transfer control. The design of these subsystems is patterned after those in HNPF, with heat-transfer requirements established by the turbine-generator, and reactor control in cascade with the heat-transfer system. This method provides system stability as well as excellent control of steam conditions.

Another important criterion in control design philosophy is control by matching sodium and water flows to power demands and by matching reactor heat rate to primary sodium flow. The speed with which flow and reactor heat-rate (neutron flux) can be changed, and the heat capacitance and transfer of sodium make the system unusually responsive to turbine generator demands. A corollary control philosophy is that continued trim on plant temperatures will be made, with setpoint programming based upon plant load, so that rates-of-change of temperature will not be excessive. As a consequence of these design criteria, reactor temperature differential (also for other heat exchangers) is

nearly constant; and temperatures decrease slightly with decreased load in order to minimize desuperheating load.

Maximum use of feed-forward (anticipation) control is made in setpoint programming; setpoints are identified by PLD (plant load demand) in the control diagram (Drawing 68). As shown by the diagram, these signals are not only set temperatures but also furnish the primary index for sodium and feedwater flows. Final control is by standard 2- and 3-element controllers. The PLD inputs to local controllers come from a unit called the plant load computer. As in HNPF, this consists of a rack of ratio-relays with bias regulators for temperatures and sodium flows, and characterizers for feedwater flow. However, alternate methods of mechanization are feasible and may appear desirable as design progresses. The input to the plant load computer is essentially the same as to the governor control, and makes maximum use of runback inputs, frequency or pressure bias, or other anticipation signals.

While control components are largely standard, it should be noted that all controllers have deviation alarms and indicators, some of which are on expanded scale. Flow and flux controllers are rate-limited and have auto-transfer stations. All subsystems can be controlled manually, with a remote manual station for important controllers.

A brief description of the response of the plant to a load increase is presented below.

- 1) The new load demand is simultaneously presented to the governor control and plant control system. This may come from area dispatch or runback inputs or may even be locally generated. Frequency or pressure bias may be introduced, but nominally the signal to the governor and plant control system differ only in requirements of turbine pressure control. These inputs determine increased setpoints for boiler feed, main and reheat sodium flow, and sodium temperatures. Local controllers then act to increase flows and temperatures. Final control of steam temperatures (to the high- and low-pressure turbines) is by attemperation, nominally only during load changes, with minimum duty at steady-state operation.
- 2) Within the boiler feed and primary sodium flow control systems, reset signals from steam and sodium temperatures act to further stabilize the process.

- 3) Within the plant control system, increased primary sodium flow calls for rod movement to increase nuclear heat rate. Trim signals based upon fast-response fuel channel outlet sodium temperature measurements act to maintain a stable heat balance and scheduled temperature control.
- 4) At the increased power level, the plant is then stabilized at the higher neutron flux and increased sodium, feedwater, and steam flows.

B. TURBINE PLANT INSTRUMENTATION

1. Turbine-generator and Main Steam Control

Conventional turbine-generator controls and supervisory equipment are provided in a centralized control room. Two of the seven extraction steam feedwater heaters are mounted in the condenser neck and do not require extraction trip valves. The five other extraction heaters are provided with extraction trip valves which all close automatically on turbine trip or individually on high level in the heater. This prevents water from entering the turbine or overspeeding the turbine from steam generated by the flashing of water in the heater. Alarms from local control boards are provided in the control room for malfunction of the hydrogen seal oil unit and the oil purifier system. The main steam flow signal for the feedwater control system is generated by the pressure-drop across the first stage of the high-pressure turbine.

2. Condensate and Feedwater Control

Feedwater control is derived from plant load demand; and process signals from steam flow, feedwater flow, and steam generator outlet sodium temperatures. Signals from the controller accomplish the following functions: (1) control of the speed of the shaft driven pump, (2) control of the flows to the individual module groups, and (3) regulation of flows when using the 25% of rated capacity startup pump. The controller may be positioned for either manual or automatic control from the control room for any phase of operation. Split-range control valves are provided to effect the best control over the wide range of operating conditions. Feedwater flow is measured by holding ring type flowmeters welded into the feedwater lines.

Conductivity measurements of the hotwell condensate are taken in each of four sections of the hotwell as well as in each of the condensate pump suction

lines to aid in isolating tube leakage in the condenser. Hotwell level control is by a conventional makeup and drawoff system, and condensate flow control is by bypass flow to the condenser. A sample connection, upstream of the demineralizer, is provided for dissolved oxygen determinations to serve as a reference for condenser oxygen removal efficiency and a guide for chemical treatment requirements.

The system demineralizer is a bypass-type unit so constructed that it is capable of polishing 90% of rated flow following a system upset. Control and instrumentation for the demineralizer is from a local board with control room annunciation of malfunction. Start-stop control is manual, except that regeneration stops automatically.

Condensate tank level is indicated locally. However, high- and low-level alarms are provided in the control room. A makeup water demineralizer is provided and arranged for manual start and automatic shutoff from high condensate storage tank level. Regeneration is manually started.

C. NUCLEAR INSTRUMENTATION (Drawing 69)

The basic function of the nuclear instrumentation system is to provide continuous measurement, indication, and recording of reactor thermal neutron leakage flux from shutdown to full power, and to generate signals calibrated to represent core power and rate of change of power for the automatic control and protective systems. Reactor leakage flux measurements are performed by sensors located in the biological shield at the reactor vessel periphery.

The subordinate function of the system is to provide core flux level measurement during fuel loading operations, and during low-power experimental operations when reactor leakage flux is below the detectable range of the sensors in the biological shield. Core flux measurements for these operations are performed by movable sensors located in thimbles inside the reactor vessel. Location of these thimbles and the peripheral thimbles relative to the core, as well as other reactor instrumentation are shown schematically in Drawing 70.

Because reactor neutron leakage flux varies over a span greater than 10 decades in going from shutdown to full power, and since no single sensor type is available with this range span, use of three types is made to provide continuous and total range coverage. These sensors divide the entire flux span

into a source range, intermediate range, and power range with at least one decade overlap between successive ranges to ensure continuous coverage. The nine channels comprising the source, intermediate, and power range nuclear instrumentation are shown in Drawing 69. The incore channels are not shown.

System design criteria are based on providing a high degree of reliability in flux level measurement for safety and control functions by use of redundant measurement channels. Startup accidents are considered potentially more hazardous than others and thus, for safety in the startup phases of reactor operation (when the turbine-generator is off the line), duplicate channels are provided in the source and intermediate flux ranges. Each channel is independently capable of initiating protective action if abnormal conditions arise, or instrument failure occurs. In startup, the consequences of false protective action are not severe, as the machine is not on-line.

During power operation, reliable flux level indication is more critical. Five identical channels are used; three for the protective system, where the signals are used in a 2-out-of-3 coincidence, and two for the plant control system where the signals are used in an auctioneer-average system. This arrangement in HNPF has demonstrated that excellent safety and control are achieved with a substantially reduced chance of plant shutdown due to instrument failures. An additional single channel is provided for independent monitoring of both the intermediate and power range. Laboratory-type readout is used, with expanded scales suitable for calibration.

During fuel-loading operations, leakage flux from the partial core is inadequate for safe operation. Two additional channels are installed temporarily in the core, and used for incore flux mapping system.

To perform the basic and subsidiary functions and satisfy design criteria, the system consists of the following channels of instrumentation.

1. Source Range Channels I and II

The identical source range channels each consist of a fission chamber, preamplifier, adjustable voltage power supply, period computer, period indicator, and level indicator. There is a source range level recorder which can be switched to either Channel I or II. The source range period signal is not used to initiate protective action; it is for alarm and operator information only.

2. Intermediate Range Channels III and IV

The intermediate range channels consist of a compensated ion chamber, adjustable voltage power supply, period amplifier, period indicator, and level indicator. Intermediate range period and level recorders can be independently switched to either channel. The intermediate range period signal can initiate setback action up to 10% power, at which point it is bypassed, since other protective action is more reliable above that power level.

3. Power Range Channels V, VI, VII, VIII, and IX

The power range channels consist of an uncompensated ion chamber, adjustable, voltage power supply, current amplifier, and level recorder. Channels V, VI, and VII provide signals for the measured neutron flux and computed outlet temperature sections of the plant protective system. Channels VIII and IX provide neutron flux signals for the plant control system.

4. Intermediate and Power Range Channel X

This channel consists of a compensated ion chamber, adjustable voltage power supply, micro-microammeter, voltage to frequency converter, scalars, and printout equipment.

5. Incore Low Power Channels XI and XII

The incore channels consist of a high temperature fission chamber, preamplifier, adjustable voltage power supply, linear countrate amplifier, and level indicator. A level recorder can be switched to any one of the three channels. The linear countrate amplifiers utilize range switching to effectively cover several decades and have provision for protective signals for use during the fuel loading period.

D. PLANT PROTECTIVE SYSTEM (Drawing 71)

The function of the reactor plant protective system is to prevent thermal damage to the reactor system in the event of accidents or plant component malfunctions. The protective system performs this function through alarm, setback (driving in all control rods for a short period of time), and scram (full insertion of all control rods by gravity fall) action. Alarm in the plant control, nuclear, protective, and radiation monitoring system is included in the reactor annunciator system with suitable color coding for ready identification. Additional

protection is provided by startup and operating interlocks that prevent withdrawal of control rods under conditions predetermined to be unsafe.

System design is based on early detection and indication of conditions which could endanger personnel or cause damage to the plant if uncorrected. Specific subsidiary criteria are as follows.

- 1) Protective circuitry independent of control circuitry.
- 2) Independent annunciation for each function with ready identification of the cause of setback or scram.
- 3) Coincidence circuitry, "hot spares," and test equipment to minimize false action.

The plant protective system subsystems can be briefly described as follows.

- 1) Manual: provides scram or setback at the discretion of the operator.
- 2) Reactor low sodium level: provides protection against low sodium level and probable loss of pump suction due to piping rupture or other malfunctions.
- 3) Primary to secondary flow ratio: provides protection against abnormal flow ratios due to malfunctions in the pumps, pump drives, control system, control valves, and the control valves, and the piping system.
- 4) Secondary to feedwater flow ratio: primarily for protection against malfunctions in the feedwater or feedwater control systems.
- 5) Computed outlet temperature: the outlet temperature is computed according to the formula:

$$\text{Outlet temperature} = \text{inlet temperature} + (\text{a constant}) \frac{\% \text{ rated power}}{\% \text{ rated flow}}$$

This computer protects against unbalanced power/flow ratios while accounting for the changes in reactor inlet temperature.

- 6) Reactor power level: protects against excessive temperatures in the fuel and other reactor components by limiting reactor power regardless of the resulting sodium outlet temperature.
- 7) Measured fuel channel outlet temperature: primarily intended as a backup for the computed outlet temperature circuit.

- 8) Rate of change of fuel channel outlet temperature: provides protection against the stresses, which result from rapid changes in sodium temperature. This circuit is sensitive to both positive and negative temperature changes; therefore, no setback action is provided since a setback would only aggravate a negative temperature transient.
- 9) Intermediate power range short period: provides protection against dangerously fast power increases during the startup operation. This circuit utilizes setback action only and is bypassed in the power range.
- 10) Turbine trip: intended to reduce reactor power in the event of a turbine trip, if not reset within a predetermined time.

Drawing 71 is functional in nature showing only those signals and actions associated with scram or setback. It indicates, for example, that both scram and setback can be generated manually, and that only one signal is required. Only setback (and not scram) is caused by intermediate range short period and only one signal of the two available is required. The drawing also shows three principal actions on setback and seven on scram. Again it is emphasized that the diagram is functional, since in two cases more than one function is included in a single equipment chassis. The high temperature and rate-of-change of temperature is one; excessive reactor power level and computed outlet temperature is the other.

Those items shown in Drawing 71 as 2 of 3 indicate that three separate computer systems exist, and that at least two of these must indicate an abnormal condition before action is taken. All alarms (as opposed to setback or scram) are actuated by a single unit, even in multiple unit systems.

E. RADIATION DETECTION AND MONITORING SYSTEM

Design of the radiation detection and monitoring system is based on protection of personnel within the plant and prevention of release of radiation from the plant in excess of safe values. The design follows that of HNPF and will consist of: (1) remote area monitoring system; (2) airborne particle monitors; (3) cell atmosphere monitors; (4) secondary sodium monitors; and (5) vent gas, prestack, and stack monitors.

In addition to main and local board annunciation, local audiovisual alarms and indications of reactor operation are given. Switching is performed automatically to prevent radioactive effluent release from the plant by diversion to holdup tanks.

F. SCANNER-LOGGER SYSTEM (Drawing 72)

The purpose of the scanner-logger systems is twofold: (1) to rapidly scan temperature signals from the reactor complex and sound an alarm if any temperature is outside the prescribed limits; and (2) to print a complete record of these temperature signals on command or at some predetermined time.

The scanner-logger will be capable of monitoring the output of 300 thermocouples. If the temperature signal from any one of these thermocouples falls outside the high and low limits (which can be adjusted separately by the operator), a horn will sound, a high or low alarm light corresponding to the thermocouple location will light, and an alarm printer will record the time, thermocouple position, measured temperature, and the high and low alarms settings at that time.

Upon command from the operator or at predetermined time intervals, a complete record of all temperatures will be made. This complete record will be completed in approximately 10 min, with the actual time printed each minute during this interval. The record for each point will consist of position identification and the measured temperature. Provision will also be made to allow the operator to individually monitor any thermocouple at any time by connection to a digital temperature indicator.

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VIII. AUXILIARY SYSTEMS

Various supporting services are required by the power station. Auxiliary systems and equipment are provided to supply these services so that the plant is entirely self-sufficient. These systems and their function include the following:

- a) Service water supply system — provides secondary water requirements such as domestic water, makeup to demineralizer, and jockey fire pump supply.
- b) Closed cooling water system — provides cooling of equipment in both reactor and turbine plants.
- c) Instrument and service air system — provides compressed air for normal services, and compressed, filtered, and dried air for instrumentation.
- d) Gas supply system — provides helium, nitrogen, hydrogen, and carbon dioxide as required throughout the plant.
- e) Loading face shield cooling system — cools loading face shield by circulation of compressed nitrogen.
- f) Cell nitrogen cooling system — maintains proper operating temperatures for equipment in the cells.
- g) Radioactive vent — collects, filters, monitors, and controls disposal of all gases.
- h) Radioactive liquid waste — segregates, stores, monitors, and disposes of plant liquid wastes.
- i) Sodium service — provides facilities for filling and draining sodium systems and for sodium purification.
- j) Component maintenance — provides facilities for the washing, disassembly, reassembly, packaging, and shipment of radioactive components.
- k) Sodium systems preheating — provides electrical preheating of sodium piping and components.

- l) Steam dump and startup system — provides steam and steam sinks for startup and shutdown, respectively.
- m) Fire protection — provides automatic detection and warning of fires, and automatic and manual fire-fighting equipment.
- n) Heating, ventilating, and air conditioning — conventional equipment supplies these services.

Each system and its function are described in more detail below.

A. SERVICE WATER SUPPLY SYSTEM

Raw makeup water is supplied from the river by two pumps, each rated at 200 gpm at a TDH of 100 ft. The river water is stored in a 200,000-gal. cone-roofed carbon steel tank. The river water storage tank supplies water to the jockey fire pump, domestic water pumps and the makeup demineralizer feed pumps.

A makeup demineralizer rated at 75 gpm is provided. Water from the storage tank is supplied to the demineralizer by two full-capacity feed pumps rated at 75 gpm and a TDH of 100 ft. These pumps are also used for backwashing. The demineralizer is supplied as a unit complete with controls.

The demineralizer effluent flows to the condensate storage tank. A condensate transfer pump rated at 50 gpm at a TDH of 75 ft is provided to supply demineralized water from the condensate storage tank for various services, including makeup to the closed cooling water system and backwash water for the condensate demineralizer.

Water from the storage tank is also supplied to two full-capacity domestic water pumps which supply water to the domestic water accumulator. The pumps are rated at 50 gpm at a TDH of 140 ft. The accumulator is a 550-gal. galvanized vessel, and contains controls similar to the fire system accumulator. A hypochlorinator is provided between the pumps and accumulator.

Two vertical, wet-pit, motor-driven centrifugal pumps supply river water for cooling the turbine plant and reactor plant closed-loop cooling water heat exchangers. Each pump is rated at 2500 gpm at a TDH of 100 ft.

B. CLOSED COOLING WATER SYSTEM (Drawing 73)

A 2-branch closed cooling water system is provided. The turbine plant branch serves the high-pressure and low-pressure seal oil cooler, vacuum pump, turbine-generator hydrogen cooler, instrument and service air compressor jackets and after coolers, heater, drain pump, startup feedwater pump and lube oil coolers, miscellaneous pump bearing cooling, sample coolers, the control room air conditioning unit and ventilation system. The reactor plant branch serves the reactor instrument thimbles, cell nitrogen cooling system, component wash cells cooling coils, R/A vent compressors intercoolers, R/A vent condenser, wash-cell vent compressor intercooler, radioactive waste transfer tank, and the reactor loading face shield compressors and coolers.

The system contains two full-capacity coolers, two full-capacity pumps and a small surge tank which floats on the suction side of the pumps. Under normal operation, makeup water is supplied from the main condensate system. For initial fill and for periods when the main condensate pumps are not in operation, makeup water is supplied by the condensate transfer pump. Buffering chemicals are added to treat the water for corrosion and scaling control.

The pumps are horizontal centrifugal motor driven pumps at 1400 gpm at a TDH of 150 ft. An automatic standby-start arrangement actuates the standby pump in event of failure of the operating pump. Two full-capacity shell and straight tube type heat exchangers are provided. The exchangers are designed for a duty of 7 million Btu/hr, and each has 5000 ft² of surface. Admiralty tubes are used in all of the exchangers. Service water from pumps at the river supply water through the tube sides for cooling. One cylindrical, vertical, carbon steel surge tank with a capacity of 2000 gal. is supplied.

C. INSTRUMENT AND SERVICE AIR SYSTEM (Drawing 74)

Three motor-driven air compressors are provided for the instrument and service air systems. Two compressors, each capable of supplying total turbine plant and reactor plant requirements, provide instrument air. One service air compressor is furnished. The instrument and service air systems are completely separate except for an emergency cross-tie to the instrument air from the service air.

The cross-tie contains an oil separator-filter and a control valve which opens on low instrument air header pressure.

Each instrument air compressor discharges through an aftercooler into a receiver. From the receivers the air passes through a dryer and a filter regulating station from which it is distributed throughout the plant. Controls are provided so that either compressor can be put on "run" or "autostart."

The instrument air compressors are Teflon ring, reciprocating, single-stage with motor drivers and V-belt drive. Each compressor is rated at 150 scfm at 125-psig discharge pressure.

Service air is supplied by one 200-scfm-capacity, oil-lubricated, motor-driven air compressor also delivering air at 125 psig.

The service air compressor discharges through an aftercooler into a receiver, and from where is distributed throughout the plant.

All three compressors are equipped with intake filter silencers and after-coolers with moisture separators and drain traps; and are designed for continuous operation. They maintain receiver pressure by means of automatic unloading devices.

Each of the three air receivers has a capacity of 35 ft³.

One dual-chamber, heater type air dryer utilizing solid desiccants is provided for the instrument air system. No heat is required for regeneration. Moisture in the entering air will be absorbed, and later will be discharged to the atmosphere with the purge air. The dryer will be a self-contained unit, complete with all valves and controls for automatic operation.

D. GAS SUPPLY

Four gases are supplied: helium, nitrogen, hydrogen, and carbon dioxide.

1. Helium (Drawing 75)

The helium maintains an inert gas blanket in the reactor and in all piping, vessels, or equipment where a free surface of sodium exists in order to prevent oxidation of the coolant. The system is designed to maintain constant gas pressures required in the reactor, primary system, the secondary heat-transfer system, and in the primary and secondary drain tanks. Helium is used as the

displacement gas during draining, filling, and transferring sodium. In addition, helium is supplied to control rod thimbles, component wash cell, fuel-handling machine, and sodium freeze seals.

The helium gas pressure in the primary sodium system is sufficient to prevent air from entering the vessels, but is low enough to minimize hazards should a leak occur. The pressure control range is such that the gas flow from the vessels through the vapor traps will be minimized to reduce gas usage and sodium vapor transport by aerosol entrainment. The gas pressure on the secondary heat removal system is of such magnitude that should a sodium leak occur in the intermediate heat exchanger, leakage would be nonradioactive secondary sodium into the primary system, rather than vice versa.

On demand, the gas passes through NaK bubblers. This system consists of three tanks in series. The first unit serves to remove any foreign material including liquid water, and also acts as a surge tank in the unlikely event of a backflow. The second allows the helium to bubble through the NaK, and the third removes any NaK entrainment. The helium will be dry and free of oxygen before it is admitted to the sodium system. If, for any reason, the NaK bubblers become inoperative, and the pressure downstream of the last vessel becomes less than 70 psig, a bypass line will open to supply the required helium gas. An alarm signals this action, indicating possible plugging of the system.

Helium is supplied from a leased helium storage tank. From the storage tank, helium flows to a 35-ft³ surge tank, where it is kept at 100 psig. Gas pressure is controlled by a pressure reducing station. Low pressure alarms are provided. Pressure relief valves are connected to the radioactive vent system to prevent over pressurization. Instruments are provided in the helium supply lines to monitor helium usage.

2. Nitrogen (Drawing 76)

The primary function of the nitrogen system is to supply and maintain an inert gas atmosphere in the primary pipe galleries, heat-exchanger cells, sodium purification and fill tank cells, and the maintenance cell to eliminate the possibility of a fire in the event of a radioactive sodium spill. The nitrogen system also provides cooling for the primary sodium butterfly valves freeze

seals, the flow meter magnets, the primary cold traps, plugging meter assemblies, reactor loading face shield, and heat transfer equipment cells. Nitrogen is supplied for purging or draining operations where helium is not required; e.g., steam generators and for operation of instruments and pneumatic tools used in cells where an inert atmosphere is required. A nitrogen atmosphere is maintained in a sump tank connected to the rupture discs on the steam generators and reheaters, in the improbable event of steam generator rupture. As the sump tank fills, liberated hydrogen cannot contact oxygen containing air. During the sodium-filling operation, nitrogen is used to maintain an inert atmosphere in tank cars. Nitrogen also serves as an emergency backup supply for the instrument air.

The nitrogen supply system consists of a liquid nitrogen storage tank, an evaporator, and a pressure control unit, packaged into an integral system and furnished by the nitrogen vendor on a leased basis. Liquid nitrogen is brought to the plant in trailers of 25,000 scf capacity each. A pressure regulator reduces the pressure to 100 psig. The 100-psig nitrogen supplies the maintenance cell door inflatable seal, and provides motive power for operating the primary sodium butterfly valves.

For low-pressure nitrogen service, the 100-psig nitrogen is reduced to 15 psig; and a header leads to the vault where the pressure is again reduced. The system used to circulate nitrogen for cooling the cells is discussed under "Cell Nitrogen Cooling System."

3. Hydrogen

A supply manifold, with provision for a supply trailer and regulating valves and pressure reducing valves, supplies gas as required to the main generators for purging and makeup. Circulating blowers and coolers are provided with the generator. An oxygen analyzer samples the circulating hydrogen within the generator.

4. Carbon Dioxide

A supply trailer bank is provided for general station fire fighting and turbine cooling purge. Nozzles are provided at the main transformer, generator, and motor-operated equipment which are activated by a rate-of-temperature-rise fire-control system. The carbon dioxide system is not used for sodium fires, and is not supplied to the nuclear portion of the plant.

E. LOADING FACE SHIELD COOLING SYSTEM (Drawing 76)

The loading face shield cooling system removes heat transferred from the upper plenum and heat generated by the attenuation of radiation from the reactor, by circulating nitrogen gas in tubing imbedded in the lead above the bottom seal plate. The cooling system is a completely enclosed and isolated system circulating nitrogen at approximately 250 psig. The system contains two cooling compressors and two heat-exchangers in parallel, each capable of handling the total cooling load. A snubber provides a sensing point for the system pressure and an expansion volume for surges from the compressor and minor system temperature variations. The circulating compressors are conventional, horizontal, reciprocating compressors with special seals to minimize leakage. The heat-exchangers are water-to-gas shell and tube type, and are cooled by the closed circuit cooling water system.

From the snubber, the nitrogen gas passes through the tube side of one of the two parallel heat exchangers where the nitrogen is cooled to about 95°F. The cooled nitrogen passes from the heat-exchangers through the coils of the loading face shield where heat from the loading face shield is absorbed by the nitrogen. Makeup to the system is supplied from the plant nitrogen system by means of small injection compressors which increase the plant nitrogen supply pressure up to the system operating pressure. These small nitrogen compressors also take any leakage from the circulating compressor seals and return it to the system to minimize makeup requirements.

There are several separate cooling circuits of 2-in. diameter tubing. These coils are laid on the bottom of the loading face shield and imbedded in the layer of lead above the bottom seal plate. These coils are brought out to the top of the shield and thence to the nitrogen circulating system. Sufficient cooling capacity is provided to allow for the failure of cooling circuits without the shield overheating. The required flow of gas is approximately 370 cfm of nitrogen.

Radiation studies of nitrogen indicate that after 1 yr of operation with no purge, the nitrogen activity will not exceed the maximum permissible concentration. The nitrogen activity will be due primarily to traces of Carbon¹⁴. The system will be purged continuously due to a slight amount of leakage. Thus, the nitrogen cooling activity is not expected to reach the maximum permissible concentration.

F. CELL NITROGEN COOLING SYSTEM (Drawing 77)

The cell nitrogen cooling system maintains proper operating temperatures for the equipment in the cells. Nitrogen was selected as the atmosphere in all cells containing radioactive sodium to minimize the hazard of a sodium leak or spill. This system also maintains the concrete shield walls surrounding the cells below 150 to 180°F. The nitrogen cooling system is divided into two separate circuits. One circuit, where the nitrogen becomes activated, cools the reactor cavity, primary pipe shaft, and the fuel storage pool. The other, where the nitrogen does not become activated, cools the following cells:

- 1) Main primary pump and IHX cell
- 2) Auxiliary primary pump and IHX cell
- 3) Primary fill tanks cell
- 4) Primary sodium service cell 1
- 5) Primary sodium service cell 2.

Two sets of cooling units are provided for each of the two cooling circuits. This permits servicing the equipment in a unit without interrupting the operation of the plant.

The cooling units consist of a nitrogen to water heat exchanger, blower, and a motor mounted inside a leak-tight pressure chamber. The cooling water supplied to the heat exchanger is from the closed-loop water cooling system. The cooling units are provided with moisture detectors to sound an alarm if a water leak occurs inside the nitrogen coolers. This reduces the possibility of moisture being carried into the equipment space by the circulating nitrogen.

Each nitrogen filled cell has inlet and outlet pressure control for maintaining a slight positive nitrogen pressure (1 in. of water to 1 psig). Cell operating pressures are set so as to prevent air inleakage into the cells, and to minimize nitrogen loss due to temperature changes and atmospheric pressure fluctuations.

A pressure control valve is provided to relieve the cell pressure should it rise above a predetermined setpoint. The cells normally vent to the building exhaust stack. A radiation sensitive vent controller monitors the effluent gases

from the cells. If radiation is detected, the gases are automatically diverted to the radioactive vent system. The reactor cavity is normally vented to the radioactive vent system.

During startup, an inert atmosphere in the cell is obtained by purging with nitrogen. Sample lines from the cells are manifolded to a local board mounted oxygen analyzer for monitoring the cell atmospheres. If the oxygen content of the cell atmosphere starts increasing during normal operation, the monitoring system will detect the condition.

A backpressure regulator controls nitrogen bleedoff in order to maintain constant pressure as the gas in the vault heats up. After a reactor shutdown, the inlet pressure regulators maintain constant pressure as the cell cools down by bleeding in additional nitrogen. These valves also permit breathing due to atmospheric pressure changes.

When it is necessary to enter the cell for repairs and inspection, the nitrogen atmosphere will be replaced with air. An exhaust duct is provided and valved so that the cell may be vented at a high rate, directly to the building ventilation system and to the stack.

In the primary sodium service cells, the nitrogen flow also provides cooling to the cold traps. Dampers are required in the inlet ducts to control the amount of gas flow through the cold traps hoods, thus regulating the coolant flowrate. Additional control is provided by means of dampers to ensure adequate cooling to the various specialized items of equipment, such as electromagnetic pumps and flowmeters.

G. RADIOACTIVE VENT (Drawing 78)

The radioactive vent system collects, filters, and monitors all gases that may be radioactive. Gases are released through the plant stack to atmosphere if the activity is below the maximum permissible concentration, or stored in decay tanks for subsequent controlled release to the atmosphere if the activity is above the maximum permissible concentration. The radioactive waste gas disposal system consists of two small compressors for normal use, two large compressors for use during purging operations, three decay tanks, absolute filters, and four radiation monitors. Each of the four radiation monitors

operates a valve arrangement that directs the gas to either the effluent stack or the decay tanks. Of the three decay tanks that are provided, one is on stream or filling, the second is in the process of decay, and the third is available for discharge to the stack. The tanks are made of carbon steel and each has a capacity of 1000 ft³. Valves and piping are also of carbon steel.

Gas with an activity level above maximum permissible concentration for direct venting is sent to the vent compressor suction tank, which serves as a surge tank to reduce cycling of the compressors. The suction tank has a capacity of 300 ft³ and is made of carbon steel.

Two sets of compressors, one consisting of two 4-cfm units for normal service and the other of two 50-cfm units for use during purging and venting operations, compress the gas to 150 psig for storage in the decay tanks. Under emergency conditions, the compressors are available for parallel operation. The compressors are of the diaphragm totally enclosed type for minimum leakage.

One of the small compressors is maintained on the line at all times when the pressure in the suction tank is greater than 11.7 psia. Should the pressure rise above a second setpoint of 12.2 psia, the second small compressor starts automatically. A pressure rise to 13.2 psia starts one of the two large compressors. At a pressure of 14.2 psia, all four compressors operate simultaneously. Should the pressure in the suction tank continue to rise under these conditions, a high pressure alarm sound at the control room.

Analysis of samples from decay tanks permits determination of time of disposal and allowable release rates. Gas is released to the stack at such a rate that the activity of stack gas, when mixed with the building ventilation exhaust air, is below the maximum permissible concentration for direct release. All vented gases leaving the building are filtered and monitored prior to entering the vent stack. Radiation instrumentation in the stack (with an alarm in the control room) gives a final check on the activity level and automatically stops any radioactive gas release until manually reset. Experience to date with the Hallam Nuclear Power Facility has shown that controlled release from the decay tanks has been required for about 1% of the gas released. The frequency of routine radioactive gas release is expected to be much less during

normal, continuous operation, when testing does not contribute to the volume of gaseous waste.

Separate vent apparatus is supplied for the wash cell, consisting of transfer tank, decay tank, vent compressor, cooler, vent condenser, and absolute filter. This apparatus is separate from the main radioactive vent system in order to prevent hydrogen, evolved during the process of washing sodium from components, from entering the vent system. Air can be drawn into the vent system during purging operations. With separate systems, hydrogen from the wash cells cannot mix with any air that may be present in the main decay tanks.

A radiation detection system is provided to monitor the reactor helium cover gas for fission products. During normal reactor operation, Xe^{133} is present in the cover gas at an equilibrium concentration due to leakage of xenon through the fuel cladding. The radioactivity level of the Xe^{133} varies as a function of reactor power and temperature. By continuously recording the cover gas radioactivity, a rapid increase in the activity level (unaccounted for by reactor operation) can be detected and indicates a fuel cladding failure.

This system consists of a scintillation-type detector, amplifiers, differential analyzer, 6-decade log countrate meter, recorder, annunciator, and power supply. A continuous sample of helium cover gas is taken by means of a circulating gas pump from the reactor vent line between the vapor trap and radioactive vent backpressure controller, passed through the radiation detector, which consists of a shielded lead chamber containing a scintillation detector, and then returned to the reactor through the helium supply line. This avoids interference with proper operation of the reactor pressure controls or leakage monitoring equipment. Since the gas sample may be radioactive, all the detection equipment will be located behind shielding. Sodium will be removed from the circulating stream by means of a vapor trap, thereby eliminating Na^{24} activity which could otherwise mask any xenon activity.

H. RADIOACTIVE LIQUID WASTE (Drawing 79)

The radioactive liquid waste system monitors, segregates, stores, and disposes of the liquid wastes. The system collects wastes from the vent compressors drain, the sodium component wash cells, and the maintenance cell.

Wash-water from the component wash cell is collected in a waste transfer tank. Waste from the maintenance cell and decontamination area is collected in holdup tanks. Radioactive gas compressors and decay tanks drains are collected in a collection tank. Samples are removed from these tanks and monitored for activity. If the activity in the transfer tank is less than maximum permissible concentration for direct release (wash-water will not be contaminated after component washing in the preparational test period, for example), the waste is transferred directly to the condenser cooling water discharge. Normally, Na which has a 15-hr half-life, is the only radioactive isotope in wash-water. If the waste activity is above the maximum permissible concentration, it will be transferred to a decay tank and discharged after the radioactivity has decayed to a permissible concentration. Dilute waste from any source with long half-life isotopes also would be transferred to one of the decay tanks and held for later disposal. Any waste containing high level, long half-life isotope contamination would be placed in the concentrated waste tank and held for disposal.

The radioactive waste transfer tank has a capacity of 1200 gal. and is made of carbon steel. There are two carbon steel maintenance cell holdup tanks, each having a capacity of 500 gal. There are also two decontamination room holdup tanks which are similar to the maintenance cell holdup tanks. The collection tank is made of carbon steel and has a capacity of 100 gal. The long half-life and short half-life tanks each have a capacity of 2000 gal. and are made of carbon steel. All tanks, except the small collection tank, are equipped with an air-sparger for mixing, a station for manually obtaining a sample before emptying the tank, liquid level indicators, and high and low level alarms audible in the control room.

The liquid waste transfer pumps are conventional centrifugal pumps equipped with a mechanical seal.

Ultimate disposal of waste containing isotopes with long half-lives is by land burial or sea disposal. The choice would be on an economic basis, which is influenced by plant location. The contaminated liquid is mixed with clay or cement for solidification and cast in steel drums for disposal. Low level waste (from a decay tank) is shipped in unshielded drums. High level waste must be shielded during shipment.

Laundry facilities are not included in the plant because they are uneconomical at most locations. Coveralls and other protective clothing used in contaminated areas can be monitored, segregated, and handled as indicated below.

<u>Activity</u>	<u>Disposition</u>
None	Local laundry
Moderate	Commercial radioactivity laundry
High	Disposal (packaging and burial)

Under normal anticipated operating conditions there should be no high-level activity laundry produced.

I. SODIUM SERVICES

Facilities are provided to fill, drain, and flush the primary and secondary sodium systems with sodium, and for purification of the sodium. The system provides a means for melting and filtering the sodium as received from tank cars, trucks, or drums. In addition, the system is capable of individually filling and draining each loop of the sodium heat transfer system and the reactor vessel. In the remote event of leakage from the reactor vessel and its liner, the reactor cavity can be drained.

Equipment provided in the sodium service system includes fill-tanks, electromagnetic pumps for sodium transfer, hot-traps for carbon removal and oxide polishing, cold-traps for sodium oxide removal, and plugging meters to determine oxide concentration. This equipment is installed in both the primary and secondary systems.

1. Fill and Drain

There are five fill and drain tanks: three 34,500-gal. tanks serving the primary system and two 25,000-gal. tanks serving the secondary system. Four 100-gpm-capacity service pumps are provided in the sodium service system. These pumps perform filling, draining, and circulation for purification prior to startup of the plant.

2. Hot-Traps

Carbon content of the sodium must be kept at low levels to ensure that at elevated temperatures stainless steel surfaces do not become carbonized

by material transport. Hot-traps are provided for carbon removal. Stainless steel is used as a sacrificial carbon getter. Flow during hot-trapping is from the discharge side of the main pumps, through the primary purification economizer, hot-trap heater, where it is heated to 1200°F, and through the hot-trap. Flow in the hot-trap is into an upper nozzle, down through a center pipe, and then upward at a slow velocity through stainless steel foil. Sodium returns to the main heat transfer system on the suction side of the pump after passing through the economizer.

The effectiveness of carbon trapping is checked by two methods. Samples of sodium are routinely taken and a chemical analysis performed. An alternate method is by installing stainless steel coupons on the outlet of the hot trap and by examining them periodically for degree and depth of carburization.

3. Cold-Traps

Cold-traps are provided to remove sodium oxide from the coolant. A typical cold-trap is illustrated in Figure 11. Flow during cold-trapping is through the primary purification economizer and through the cold-trap. A flow of nitrogen from the nitrogen cell cooling lowers the sodium temperature in the cold trap. Since sodium oxide solubility in sodium decreases with lower temperatures, the sodium oxide precipitates out in the cold-trap.

4. Plugging Meters

Plugging meters are provided to indicate sodium oxide content. Sampling connections are provided from the main sodium service system on a bypass downstream of the plugging meters. This location was selected because plugging meters will generally be in continuous operation while other parts of the service system may not be operating at all times.

The plugging meter consists of a preliminary regenerative type heat exchanger; a cooler fed by nitrogen from the nitrogen cooling cell; and a plugging valve, together with the necessary controls. In operation, sodium flow is initiated through the unit at approximately 1 gpm. The sodium is gradually cooled by nitrogen down to the point at which flow suddenly drops and then stops. These points are recorded on a recording chart showing both temperature and flow. This plugging temperature indicates the sodium oxide content of the coolant. When flow drops below a predetermined minimum, a timer automatically opens the plugging valve, stops cooling, and initiates heating in preparation

Figure 11. Circulating Cold-Trap

for the next run. Time cycle depends on the oxide content of the sodium. It is expected that a plugging run will be on the order of 30 to 60 min.

5. Location of Sodium Purification Equipment

There are two purification equipment cells for the primary system. Each cell contains a hot-trap, a cold-trap, and a plugging meter assembly. One of the two cells holds a 100-gpm sodium service pump.

The plugging meter, sodium sampler, cold-trap, hot-trap, and associated equipment in the secondary system, are in essence identical with those in the primary system, except that they are not radioactive; and hence, will be somewhat simpler in design and construction, as remote handling is not necessary.

J. COMPONENT MAINTENANCE

1. Maintenance Cell (Drawing 80)

The maintenance cell is designed for the disassembly, reassembly, packaging, and shipment of radioactive components such as fuel elements control rods, pumps, and the internals of the fuel-handling machine. It consists of two areas, designated the operating area and in-cell area.

The operating area is divided into three operating levels. Hatches are provided at each operating level to allow removal or interchange of equipment between levels. Viewing windows to the maintenance cell are provided at all three levels. A section of each of the two lower operating levels is reserved for travel of the fuel shipping cask between the cask hatch in the reactor room floor and the cask in-cell access plug of the maintenance cell. The cell roof is a rectangular plug that will allow removal of the in-cell crane as a unit.

A tunnel access door seals off the cell from the tunnel that leads to a hatch in the decontamination room. This door incorporates a folding shelf which is used for conveyance of tools or equipment to and from the decontamination room.

Located within the cell at the lower level is the vent filter installation. This installation is remotely changeable, and it will be serviced in its entirety from within the cell.

The maintenance cell is equipped with the following major pieces of equipment.

- a) An in-cell hoist capable of providing full coverage.
- b) Shielding viewing windows located as stated above.
- c) A pair of heavy duty master-slave manipulators which can be mounted through the operating face at any one of the three operating levels.
- d) A tool positioner mounted on vertical tracks capable of being positioned at any cell depth by a ball screw.
- e) A remotely operated servo-manipulator which rides on vertical tracks. The manipulator carriage can also be positioned at any cell depth by a ball screw.
- f) An audio system with a microphone located at three different vertical locations in the cell, and speakers located at all three operating levels. The system can be controlled so that sounds within the cell can be located and transmitted to any one of the operating areas.
- g) An in-cell radiation monitor which consists of a beta and gamma detector of the ion chamber type, and portable readout equipment in the operating area.

Special tools are designed for use in the maintenance cell. These and other standard tools are stored on the wall space directly in front of the viewing window.

The cell is provided with a combined ventilation and inert gas system. When any of the ports or access doors are opened or when operations involving no hazards are to be performed in the cell, a high-volume ventilation system provides the cell with an air atmosphere with sufficient flow to ensure that all air motion is inward rather than out to the work areas. When irradiated fuel elements are handled in the cell or when cell operations involve a fire hazard, a low volume vent and exhaust system and an inert gas system maintain an atmosphere of nitrogen in the cell. Under these conditions the cell is maintained at a negative pressure and the oxygen content is limited to 2%.

A cooling system is provided to limit the maximum ambient operating temperature in the cell to 150°F. Exhaust from the cell is passed through in-cell filters and directed through final absolute filters to either the holdup decay tanks or central ventilation exhaust system, as required by the level of activity of the gas.

2. Wash-Cell (Drawing 81)

The wash-cell is provided to remove sodium from components requiring maintenance. The cell consists of a carbon steel-lined cavity into which are inserted removable inner cells. One of the inner cells is made of carbon steel coated with caustic and radiation resistant paint, for cleaning items such as valves or sections of sodium piping. The other inner cell is made of stainless steel and is used to wash small core components, such as thimbles and control rods.

All cell operations such as washing, drying, and purging are performed and controlled from a central operating board which contains the necessary valves and instrumentation. Steam, water, nitrogen, helium, vacuum and vent services are provided to the wash-cell. Steam and contaminated materials washed from components pass through the radioactive liquid waste drain piping to a combination direct-contact condenser and transfer tank.

Radiation shielding for the top of the wash-cell cavity is provided by steel-encased, stepped concrete plugs. The space allowed for piping connections is filled with steel shop. Below the plug, a steel plate serves as a seal to contain the shop just below and around the piping connections and between the wash-cell and the cavity liner. This arrangement facilitates insertion and removal of the liners.

K. SODIUM SYSTEMS PREHEATING

The entire sodium system must be heated prior to initial filling of the system with molten sodium and for maintenance of the sodium in a fluid state prior to the start of nuclear power generation. To accomplish this, the sodium system is preheated and maintained at a temperature of approximately 350°F by the use of tubular type, electrical resistance heaters installed on the piping, and underneath the insulation. Components which are thus preheated include

the reactor, the intermediate heat exchangers, sodium pumps, expansion tanks, fill and drain tanks, steam generator and sodium system valves in addition to all sodium piping. The reactor core is preheated with approximately 40 removable heaters as well as fixed heaters attached to the containment vessel. The removable heaters are only required when preheating the core from room temperature. The steam generators are heated with boiler feedwater and steam.

The bulk of the electrical preheating system is operated at 480 volts, 3-phase. Individual heaters are Y-connected, resulting in an operating voltage of 277 volts. Derated in this manner, each heater will have a wattage output of one-third its rated value, and thus the heater life will be greatly extended. The preheating load does not significantly affect the auxiliary power maximum demand on the plant, since the bulk of the preheating load occurs prior to starting main pumps. The coolant circulating pumps when operating themselves contribute appreciable heat to the system.

Chromel-alumel thermocouples are utilized to monitor the preheat temperatures throughout the system, and to automatically control the temperatures where required. The thermocouples are the skin type and are heliarc welded to the pipe or component surfaces.

Sodium piping heaters, in general, are controlled by circuit breakers in the heater power panels. The heater installation will be designed to provide a steady-state pipe temperature of about 450°F, if they are left on full time. The heaters will be on only during preoperational testing and reactor startup except on certain sodium service system components. During nuclear power generation the heaters will be turned off, except for certain vent and drain lines.

Major sodium components, such as the reactor, the sodium pumps, and the sodium piping of steam generator, are preheated with automatically controlled heaters. Such an automatic control system will utilize setpoint controllers and magnetic contractors.

Multipoint temperature recorders are used to record temperatures throughout the sodium system. Thermocouples are located throughout the sodium system so that any malfunction of equipment, or abnormal operating condition of the preheating system is detected and annunciated. A total of approximately 800 recorder points is required.

L. STEAM DUMP AND STARTUP SYSTEM (Drawing 82)

The steam dump and startup system is entirely independent of normal plant hardware to allow turbine plant maintenance immediately after shutdown. In the proposed system, the emergency power loads are minimized. The system operates on principles proven successful for fossil-fired units that use a bypass flash-tank between the low-temperature and high-temperature modules. Sodium and feedwater flows are compatible with convection flow requirements for decay heat removal. Reactor power is not required to control temperature transients.

On a turbine trip the main boiler feed pump is shut down and the startup and steam dump system is placed in operation. Steam generated in the low-temperature modules is routed to the flash-tank. Flash-tank steam is then routed to the superheat and reheat modules and the atmospheric boiler. Flash-tank steam also supplies steam to the startup deaerator which is provided to deaerate the oxygen-saturated water from the condensate storage tank that has passed through the plant-polishing demineralizer. The condensate is pumped by a condensate pump coupled to a boiler feed pump with a noncondensing turbine driving both pumps. Steam for this turbine is taken from the flash-tank and discharged to the deaerator. The boiler feed pump takes suction from the deaerator and discharges to the low-temperature modules through a module feedwater heater. Steam to heat this water is taken from the flash-tank and the heater drains are routed to the deaerator.

For preheating, the atmospheric boiler is out of service, and steam to heat the circulating water is supplied to the flash tank by a fossil-fired auxiliary boiler. Water is circulated by the turbine driven pumps through the low-temperature, superheat and reheat modules and returned to the deaerator via the flash-tank. The water is heated in the deaerator and the module feedwater heater at controlled rates to establish preheat temperatures. Once the modules are preheated, water flow is discontinued and steam from the flash-tank is used to maintain temperatures using essentially the reverse of the water flow path. Steam traps are provided on the module headers to drain collected condensate to the deaerator.

During decay heat removal, when steam generation rates are too low to operate the turbine driven pump and the auxiliary boiler is not available to assure feedwater flow to the steam generators, electrically driven pumps of low capacity are required.

M. FIRE PROTECTION

1. Warning System

A fire warning system is provided to initiate an alarm in the case of fire in any area subjected to a fire potential and not under continuous supervision. The fire warning system shall consist of spot detectors, an annunciator system in the main control room, and interconnecting wiring in conduit. The detectors are actuated either by a fixed temperature rise, or rate of temperature rise, or a combination of both. Smoke detection units are also located in various areas of the reactor building, and a smoke detection system samples the air at various points in the H&V exhaust system. All wiring for the fire alarm system shall be installed within the requirements of the National Electric Code for signal-system circuits.

2. Turbine Plant

Normal fire protection will be provided by (a) a fixed fire protection system, (b) supplemental portable extinguishers, and (c) special fire prevention design features incorporated into the plant systems and equipment in potentially hazardous areas.

The fixed fire protection system provides a water spray system (manually controlled from the control room and the deluge area) in the area of the turbine generator to cover such items as the oil storage reservoir and purification system, the hydrogen seal oil unit and the confines of the sump, and any dykes surrounding the area under the turbine and auxiliary equipment. The station service and power transformers are protected by an automatic deluge water spray system. Readily accessible hose lines and reels will be distributed throughout the station so that all areas within the station with the exception of areas containing sodium are within 20 ft of a nozzle when attached to not more than 75 ft of hose. Yard hydrants and related hose stations will be strategically located throughout the yard area, and at least two fire hydrants will be installed in the outdoor substation.

Fire water is supplied by a vertical wet pit-type centrifugal pump driven through an angle gear by a diesel engine. The pump is located at the intake structure and is rated at 1000 gpm at a TDH of 290 ft. The diesel engine automatically starts on low fire system pressure. The screen wash pumps supply an emergency backup for the fire pump.

Fire system pressure is maintained at a minimum of 50 psig by a water pressure system consisting of a jockey pump and accumulator tank. The tank is galvanized steel and the correct ratio of air and water is automatically maintained in the tank by a water controller. The jockey pump discharges into the accumulator and its motor driver is controlled by a pressure switch located on the accumulator. The discharge line from the accumulator feeds directly into the fire loop. The jockey fire pump is rated at 50 gpm at a TDH of 175 ft.

An underground piping loop with weather-protected hose reels provides hose coverage to the outside areas of the plant, and in addition, hose reel stations are provided at suitable locations inside the turbine building warehouse and administration buildings. An automatic sprinkler system is provided for the main and auxiliary transformers.

A 3-ton low-pressure CO₂ system complete with controls is provided. This system is connected to the generator, the lube oil reservoir, and the lube oil filter. Portable CO₂ and dry chemical extinguishers are also provided throughout the turbine building, administration, and warehouse buildings, as required.

Special design features are incorporated in systems such as those handling hydrogen or oil, and special drainage provisions are made in areas under equipment such as transformers that could collect oil from leaks and become a fire hazard.

3. Reactor Plant

All piping and process equipment containing potentially radioactive sodium are sealed within cells and pipeways containing a nitrogen atmosphere. For small fires involving accidental sodium spills, bulk-dry calcium carbonate applied by handscoops is provided. There is no fire water in the reactor building. For larger sodium fires which cannot be approached, hand and wheeled Ansul Chemical Company, Met-L-X extinguishers are provided.

N. HEATING, VENTILATING, AND AIR CONDITIONING (Drawing 83)

1. Reactor Building

The heating and ventilating system for the reactor building consists of a once-through system, continuously circulating outside air through the building.

The air circulation rate is 200,000 cfm, which is sized to remove the heat generated within the building and to maintain personnel comfort. Outside air is filtered, heated as required, and distributed to the various operating areas. Three half-size supply fans are provided, each rated at 100,000 cfm, with one fan on standby. Air is exhausted from the building, by three similarly sized fans, to the stack.

In order to ensure that leakage of air is inward toward areas of potential radioactivity, the building atmosphere is maintained at a negative pressure relative to outdoor barometric pressure. The various operating areas within the building are maintained at differing pressures, from 0 to 1/2 in. water, in order to control the flow of air from those areas of lower potential radioactivity to those of higher potential activity. Pressure control is maintained by automatically operated dampers located in the exhaust ducting from each operating area.

All air exhausted from the building is continuously monitored for activity before release from the stack. In the event of activity levels higher than acceptable, the main supply and exhaust fans are automatically shut down and the system is isolated from the outside by closing butterfly valves located in the main supply and exhaust ducting.

All gaseous effluent from areas which could be contaminated with radioactivity is exhausted through the building stack. The minimum dilution of gaseous effluent, at the normal 200,000 cfm ventilation rate, is 10^3 at ground level (based on Pasquill's Type A weather parameters). Therefore, any radioactive gases discharged through the stack during normal operation must be at concentrations no greater than $10^3 \times$ maximum permissible concentration (MPC) allowed by Table II of 10CFR20. Any gases of higher activity are stored for decay until they can be released at a controlled rate commensurate with the above.

When the main heating and ventilating system is shut down, negative pressures are maintained within the building by a filtered exhaust system. The capacity of the filtered exhaust fan is 12,000 cfm, which is sufficient to overcome all building leakage and thus maintain negative internal pressure. Air exhausted by the fan is filtered by roughing, high efficiency and charcoal filters before discharge to the stack.

The maintenance cell is cooled by a small, recirculating, air conditioning unit located within the cell. Both the maintenance cell and the radioactive waste areas may be exhausted by means of either the main heating and ventilating system or the filtered exhaust system, depending on the activity levels of the atmosphere in these areas.

2. Turbine Room

Heating will be supplied by steam-type unit heaters. Roof exhaust fans will supply ventilation air by pulling air in through windows and doors.

3. Control Room

The control room is provided with year-round air conditioning unit for controlling temperature and humidity as necessary for proper operation of the instrument systems and for operator comfort. The unit contains refrigerated cooling coils for cooling, steam coils for heating, a supply fan, and an exhaust fan. Air conditioning is also provided to the health physics counting room and to the shift supervision office.

4. Administration Building

An overhead duct system supplies year-round conditioned air from a self-contained air conditioner. The unit will incorporate both heating and cooling coils and will provide proper outside air for ventilation. Control will be automatic.

The entire heating and ventilating system is essentially a standard industrial installation; available commercial components are used throughout.

IX. ELECTRICAL SYSTEM

A. DESIGN PHILOSOPHY

The electrical system is similar to that found in a conventional plant of the same size. The selection of equipment and material conforms to modern central station practice. Generation voltage is 20 kv, which is stepped up to the 138-kv transmission voltage by a 250-mva main transformer.

The primary plant auxiliary electrical system is 4160 volts grounded, normally supplied from the generator bus through the 12,000-kva unit station service transformer. The startup power source, a 12,000-kva reserve station service transformer, is also used for plant power supply when the generator is out of service.

The low-voltage auxiliary power system is 480/277 volts, supplied from the 4160-volt system through two 2000-kva transformers. Sectionalizing of buses, dual power feeds, and automatic transfers are provided consistent with the overall reliability requirements of the nuclear plant.

Emergency power systems are provided to serve those loads which require a high order of power continuity in order to assure either continuous operation, a safe and orderly shutdown, and protection of equipment and personnel.

B. OPERATION

The electrical system is monitored and controlled from the electrical control boards in the central control room. The plant will be started by use of the reserve station service transformer, supplied from the 138-kv outdoor substation. After the generator has been brought up to speed and synchronized with the system, the unit station service transformer is placed in service.

Motors will be controlled either from the main control room or from individual control centers at appropriate locations in the plant. All switchgear, motor control centers, and control points will be located in areas that are accessible during reactor operation at full power.

C. GENERATING UNIT

The generator rating is 210 Mw, 85% power factor, 22 kv, 3-phase, 60 cycle, 0.64 SCR, with 45-psig hydrogen. It is directly connected to the turbine shaft and

will operate at 3600 rpm. The generator leads are brought out the bottom of the stator through six bushings. Each bushing has two current transformers each for metering, relaying, and control. Suitable current transformer taps are provided for connecting the metering and control equipment. The generator is grounded through a grounding transformer to limit short circuit current to ground.

Generator excitation is provided by a suitably rated 375-volt, d-c, air-cooled exciter driven by the generator shaft through reduction gears. Excitation switchgear includes a field circuit breaker, discharge resistor, motor operated field rheostat and necessary control and protective devices, all housed in metal clad switchgear.

A spare electric motor driven exciter is provided. Its electrical characteristics match those of the shaft driven exciter, with a rating of approximately 900 kw. Synchronizing equipment is provided to permit an exciter exchange without a unit shutdown. The spare exciter is used during startup. A switching arrangement is provided which permits the use of the voltage regulation equipment on either of the two exciters.

Generator protective relays include high-speed differential, field failure, overcurrent, backup impedance, and neutral overvoltage.

Insulation will be Class B. Ratings and temperature rise will be in accordance with latest ASA Standards.

Forced air cooled, isolated phase bus is provided from the generator terminals to the low voltage terminals of the main transformer. Potential transformers and generator surge protection equipment is connected to taps to the isolated phase generator leads.

An isolated phase tap bus is provided from the main transformer bus to the unit station service transformer.

D. TRANSFORMERS

The main transformer is a 3-phase, outdoor type, Class F0A transformer, rated 250,000 kva. The transformer is protected with integrally mounted lightning arresters on the high voltage side.

The reserve station service transformer is an OA/FA, 10,000-kva, 3-phase, two winding unit, fully insulated, 138-kv delta connected on the high side.

The low side is 4160-volt wye connected, with low resistance grounding to limit the short circuit currents. Two 2-1/2% no-load tap changers above and below rated voltage are provided for voltage adjustment. Tank-mounted, station-type lightning arresters are provided on the high voltage side for protection of the high voltage winding.

The unit station service transformer is an OA/FA, 10,000 kva, 3-phase, two winding unit, 20 kv delta, directly connected to the generator bus. The secondary is 4160-volt wye connected, resistance grounded. Two 2-1/2% no-load taps above and below rated voltage are provided on the low voltage side.

The two low-voltage power transformers are 2000 kva, indoor type, askarel filled, air-cooled, 4160-volt delta on the high side and 480 volt/277-volt wye, solidly grounded on the low side. A no-load tap changer provides two 2-1/2% voltage adjustments above and below rated voltage.

The lighting transformers are 480 to 208/120 volt, 3-phase, 4-wire-grounded, air-cooled, dry tape. They may be wall- or floor-mounted as required.

E. SWITCHGEAR AND CONTROL CENTERS

The 4160-volt switchgear is one unit, indoor type, with air circuit breakers. Normally, power is supplied by the normal station transformer. Startup and alternate power are provided by the reserve station transformer. Motors, 300 hp and over, and the two 2000-kva auxiliary transformers are supplied directly from the 4160-volt switchgear.

Meters, relays, controls, and other accessory equipment required for operation and protection of the transformer and motor circuits are furnished with the switchgear. There are provisions for extension of the switchgear as required.

The 480/277-volt load centers are indoor metal clad type. They are furnished in two sections with a 2000-kva transformer connected to each section by a throat connection. A bus tie breaker is provided to connect the two sections together. Either of the 2000-kva transformers have sufficient capacity to supply the normal loads of both buses. The 480/277-volt motor control centers and motors from 125 to 299 hp are supplied directly from the bus. Meters, relays,

controls, and other accessory equipment required for operation and protection of the equipment connected to the load-centers are provided with the load-centers.

One of the load-centers is designated as the emergency bus and supplies the essential loads. A diesel generator supplies power to this bus in event of failure of both the normal and alternate power supply. All nonessential loads are automatically tripped off the bus when this occurs.

The motor control centers are metal-clad, free-standing, indoor type. Power to the motor control centers is supplied by the 480-volt load centers. The motor control centers supply power to motors from 3/4 hp up to and including 100 hp, lighting panels, and other 480-volt loads. They are located as near as possible to the loads they serve. Motors are controlled by combination magnetic starters with circuit breakers. All starters utilize 120-volt control power.

The d-c switchgear is rated 250 volts and includes air circuit breakers, motor starters, instruments, and relays. All components operate satisfactorily over the range of 210 to 280 volts dc. The circuit breakers have sufficient interrupting capacity to interrupt a short circuit at the dc bus.

F. EMERGENCY POWER SYSTEMS

Certain of the electrical loads associated with the reactor require power continuity of the highest order to ensure continuous operation, a safe and orderly shutdown, and the protection of personnel as well as equipment. It is the purpose of the emergency power system to provide power to these critical loads with the varying degree of reliability required. Table 10 lists the emergency power requirements.

1. Diesel-Generator

Emergency power is provided by a 750-kw, 480-volt, 3-phase, 60-cycle, 0.8 P. F. diesel generator. The diesel unit starts automatically upon loss of normal power. The diesel start is initiated by an undervoltage relay on the main 480-volt bus. This relay has inverse characteristics which prevent a diesel start on a transient voltage dip. The diesel-generator is capable of attaining normal speed, with a stabilized voltage output, in 6 to 10 sec.

An interruption of normal power scrams the reactor. Electrical loads which must be in service immediately following a reactor scram are picked up automatically when the diesel-generator comes into service. Other emergency

TABLE 10
EMERGENCY POWER REQUIREMENTS
(kw)

Items	Time After Scram (hr)				
	0	1/2	1	4	24
Reactor instrumentation	3-1/2	3-1/2	3-1/2	3-1/2	3-1/2
Emergency lighting	15	15	15	15	15
Radiation monitors	2	2	2	2	2
Power control circuits*	1	1	1	1	1
Fuel-handling machine*	0	15	15	15	15
R/A vent fans		6	6	6	6
N ₂ cell coolers		60	60	120	120
Main cooling water pump		120	120	120	120
Loading face shield cooling compressors		30	30	30	30
R/A vent compressors		12	12	12	12
Reactor building filtered exhaust	15	15	15	15	15
Recharge batteries		10	10	5	0
Emergency feedwater electrical pump				15	15
Instrument air compressors	30	30	30	30	30
Emergency dump system hot well pump	8	8	8	8	8
Auxiliary primary cooling pump	40*	40	40	40	40
Auxiliary secondary cooling pump	40*	40	40	40	40
Elevator	30*	0	0	0	0
Total Load	184	407	407	477	472
Size for	200	425	425	500	500

*Intermittent load

loads are brought into service manually as required. These additional emergency loads are a function of the duration of the power outage.

2. Station Battery

The noninterruptable and the noise-sensitive portions of the emergency load must be fed from a failure-free power supply. This requirement is filled by a 250-volt d-c station battery, rated 280 amp-hr at an 8-hr discharge rate. The battery cells are of the Plante positive-plate type.

The battery is capable of supplying power to the d-c loads for a period of 1 hr in the event of a failure of the diesel-generator to start. This provides sufficient time to shut the plant down in a safe and orderly manner.

3. Motor Generators

Two 120/208 volt, 3-phase, 60-cycle motor generator sets are provided to furnish noise-free precisely regulated 120-volt a-c power for the nuclear instrumentation. One of the 25-kva mg sets is capable of supplying the full load; the second unit serves as a standby.

The motor generator sets are the 3-unit type, which consists of a synchronous drive motor, an a-c generator, and a d-c motor-generator, all mounted in-line.

When normal plant power is available, the synchronous motor drives the a-c generator, which generates the noise-free 120/208-volt power, and the d-c generator, which supplies d-c power to the d-c bus and charges the batteries. On loss of normal power, the d-c generator acts as a motor and continues to turn the a-c generator. Flywheels on the shaft prevent any sudden change in speed during such a transfer of driving power.

During normal operating conditions, the noise-free power is in synchronism with the plant power. When operating under emergency conditions, the speed of the d-c drive motor is automatically regulated to the extent that the generated output has a frequency of 60 cycles $\pm 1\%$.

G. MOTORS

Motors rated 300 hp and above are fed from the 4160-volt switchgear. Motors 299 hp and below are fed from the 480-volt system. Motors of 1/3 hp and smaller are rated for 115 volts, single phase, 60 cycles. Outdoor motor

enclosures are weather-protected NEMA Type II, if available, or totally enclosed fan-cooled. Weather-protected motors are equipped with space heaters.

In general, drip-proof motor enclosures are standard for all motors located in normally clean and dry locations. In areas where conditions require, splash-proof or totally enclosed motors are used.

All motors will conform to the latest applicable NEMA and IEEE Standards.

H. LIGHTING

Lighting is in accordance with the latest recommendations of the IES as a minimum. High-bay mercury vapor lighting is furnished in the turbine room and other locations where high-bay lighting is applicable. Fluorescent lighting is provided in the control room, offices, laboratories, and similar areas. Incandescent lighting is furnished in general operating areas. Fluorescent lighting operates at 277 volts, and incandescent at 120 volts.

Emergency lighting is supplied from the 120/208-volt critical panel.

I. RACEWAYS AND WIRING

Cables for 4160-volt service are single conductor, copper, 5 kv, shielded, with butyl or oil base insulation and neoprene jacket. Power cables for 480-volt service are copper conductor, type RHW rubber or type THW thermoplastic insulation, single conductor.

Control wiring is rated 600-volt, multiconductor, with polyethylene insulation and PVC jacket over each conductor and a PVC jacket overall.

Minimum wire sizes are No. 12 AWG for power wiring and No. 14 AWG for control wiring.

All power and control wiring is routed through either rigid steel conduit, flexible metal conduit, or hinged cover steel wireways.

J. GROUNDING

A low-resistance grounding system of 1 ohm or less is installed throughout the plant. This system provides facilities for grounding all structures and equipment. Parallel paths to ground are provided for all major pieces of equipment where heavy ground-faults might occur. The main grid is 500 MCM bare copper and 1/4- by 2-in. copper bar. A complete loop of 500 MCM, bare copper

cable is buried underground around the main plant and adjacent outdoor structures. Inside the building, the main grounding grid is connected to the building steel and electrical equipment by means of 500 MCM, 4/0, No. 1 and No. 6 bare copper cables. "Cadweld" type connections are used for all grid connections. Bolted connections to equipment are made with suitable pressure-type lugs. All equipment and building steel is grounded in accordance with the National Electric Code.

K. COMMUNICATIONS

The plant communications system is an industrial type using telephone handset stations with amplifiers and speaker assemblies. Paging and party line facilities are provided.

Tone oscillator equipment is provided to sound emergency alarms over the paging system. These alarm signals are controlled from the control room.

A conduit system is provided for a telephone system to be installed by the local telephone company. Sound-powered telephone jacks are provided at each control panel together with an interconnecting wiring system.

For communications outside the plant, the local telephone company provides an automatic telephone exchange with telephones located in the control room and all principal offices.

L. ANNUNCIATORS

An annunciator system for annunciating abnormal conditions is provided for the reactor, demineralizer, radioactive waste, power generation equipment, electric auxiliaries, and power transmission equipment. The annunciators are back-lighted windows located in the control room near the associated controls for the various pieces of equipment. Audible alarms are provided for critical equipment. Acknowledge, reset, and test push-buttons are provided.

X. PIPING, VALVES, AND INSULATION

Design, material, fabrication, and installation of all piping and accessories comply with all applicable requirements of the ASME Power Boiler Code and the ASA Code of Pressure Piping, and the applicable rulings of the nuclear piping code cases to meet boiler inspection and insurance company requirements for inspection and insurance.

Adequate vents, drains, traps, and flushing connections are provided for all lines and equipment.

Gate valves have bypasses where and as required by the ASA Code for Pressure Piping. Pneumatic control valves are furnished with handjacks unless a separate manual bypass valve is provided.

Where essential to plant normal operation, valves located in inaccessible areas are made remotely operable by extending valve stems through shield walls, by use of reach rods, or by pneumatic or electric operators.

A. PIPING AND VALVES

1. Sodium and NaK Systems (including vapor lines upstream of vapor traps)

Design Conditions:	100-psig pressure, 1150° F temperature
Pipe: 8 in. and larger:	Welded stainless steel, ASTM A358 Grade TP 304. Plate per ASTM A240 Grade 304. 1/4-in. wall for pipe sizes, 8-12 in. included 5/16-in. wall for pipe sizes, 14-20 in. included 3/8-in. wall for pipe sizes, 22-28 in. included 7/16-in. wall for pipe sizes, 30 in. and 32 in. 1/2-in. wall for pipe sizes 34 in. and 36 in.
6 in. and smaller:	Seamless stainless steel, ASTM A376 Grade TP 304, Schedule 40S.
Fittings: 8 in. and larger:	Welded stainless steel, ASTM A403, Grade WP304, butt-weld ends; wall thickness to match joining pipe.
6 in. and smaller	Seamless stainless steel, ASTM A403, Grade WP304, Schedule 40S, butt-weld ends. All joints will be butt-welded.

Thermal shock relieving joints are provided wherever fluid temperature difference exceeds 150°F in joining pipe lines.

The root pass of all welds is made by the inert gas tungsten arc process without filler metal added unless of the consumable insert type. Inert gas backup of helium or argon is maintained throughout welding.

Shielded metal arc or the submerged arc process may also be employed for all other welding. Dye penetrant inspection is required for all root passes. All completed pressure containing welds will be 100% radiographed according to the provisions and standards of acceptance of Paragraph 102, ASME Boiler and Pressure Vessel Code, Section 1.

All sodium piping between the reactor vessel and the reactor cavity penetration seals, and between the seals and valve station in the sodium return lines, are provided with a guard pipe enclosure. This enclosure is made of 3/16-in.-thick plate or pipe material.

All sodium piping material, pipe and fittings enclosed by the guard pipe will be ultrasonic ally tested.

2. Inert Gas System and Radioactive Waste Disposal Systems

Design Conditions:	150-psig pressure, 200°F temperature
Pipe: 2 in. or larger:	Seamless carbon steel, ASTM A53 Grade A or B, Schedule 40,
1-1/2 in. and smaller:	Seamless carbon steel, ASTM A106 Grade A or B, Schedule 80
Fittings:	
2 in. and larger:	Seamless or welded carbon steel ASTM A234, Schedule 40, butt-weld ends.
Joints:	
2 in. and larger:	Butt-welded.
1-1/2 in. and smaller:	Socket-welded, except connections to screwed equipment or valves. Screwed connections to be seal-welded.
Valves:	
2 in. and larger:	Globe, 150-lb Y pattern, cast carbon steel body, butt-weld ends, cast-iron bolted bonnet, neoprene diaphragm and seal and disc insert.

1-1/2 in. and smaller: Globe, 400-lb brass body, screwed ends, bolted bonnet, neoprene diaphragm and flat faced neoprene disc.
Check, 300-lb brass body, screwed ends, steel seat, Buna N disc or "O" ring seat seal, spring-loaded disc.

3. Cooling Water System and Service Air System

Design Conditions: 150-psig pressure, 150°F temperature

Pipe: 2 in. and larger: Seamless or welded carbon steel, ASTM A53 Grade A or B, Schedule 40.

1-1/2 in. and smaller: Seamless carbon steel, ASTM A106 Grade A or B, Schedule 80.

Fittings:

2 in. and larger: Seamless or welded carbon steel ASTM A234, Schedule 40, butt-weld ends.

1-1/2 in. and smaller: 3000-lb screwed forged carbon steel, ASA B16.11 ASTM A105, Grade II.

Joints:

2 in. and larger: Butt-welded, except connections to flanged equipment or valves.

1-1/2 in. and smaller: Screwed.

Valves:

2 in. and larger: Gage and globe, 150-lb cast carbon steel body, flanged ends, bolted bonnet, O. S. and Y, stainless steel trim.
Check, 150-lb cast carbon steel body, flanged ends, bolted cap, stainless steel trim, swing disc.

1-1/2 in. and smaller: Gate and globe, 200-lb bronze body, screwed ends, union bonnet, I. S. R. S. bronze or alloy trim.
Check, 200-lb bronze body, screwed ends, screwed or union cap, bronze or alloy trim, swing disc.

4. Instrument Air System

Design Conditions: 150-psig pressure, 150°F temperature.

Pipe: 2 in. and smaller:	Seamless carbon steel, ASTM A53, Grade B, galvanized, Schedule 40
Pipe fittings:	
2 in. and smaller:	Screwed, 3000-lb forged steel, ASTM A105, Grade II galvanized.
Tubing:	
1/4 in. OD	ASTM B-68 copper tubing, soft-annealed, minimum wall 0.030 in.
3/8-in. OD	ASTM B-68 copper tubing, soft-annealed, minimum wall 0.032 in.
Tubing fittings:	Parker Triple-Lok brass tube fittings, or equal.
Multiple copper tubes:	ASTM B-68, Type DHP 1/4 OD x 0.030 wall, contained in an extruded virgin polyethylene sheath 1/16-in. thick, dull black, Dekoron "Metl-Cor" or equal.
Valves:	
2 in. and smaller:	Gate and globe, 150-lb brass body, screwed ends, screwed or union cap, I.S.R.S., alloy trim. Check, 200-lb bronze body, screwed ends, screwed cap, alloy trim, swing disc. needle, 3000-lb 12% chrome body, screwed ends, union cap, I.S.R.S., alloy trim.

5. Nitrogen Cooling System

Design Conditions:	This section covers all piping and ductwork, larger than 6 in. diameter, used to circulate nitrogen gas in the nitrogen cooling system.
Pipe and fittings:	All nitrogen lines, other than those within the confines of the steel lined cells, are fabricated from standard, welded, carbon steel pipe, ASTM A53 Grade A or B, minimum wall thickness as follows: <div style="margin-left: 40px;">5 1/5 in. for sizes 20 in. and larger 1/4 in. for sizes smaller than 20 in.</div> All joints will be butt-welded. All pipe buried or cast in concrete is protected with a coat of coal-tar primer followed by coal-tar enamel.

Ducts: All nitrogen lines inside the steel lined cells will be fabricated of ducts. All ducts are galvanized sheet metal, designed in accordance with the latest ASH and VE guide.

Valves: All valves located in the piping are standard ASA 150-lb flanged cast steel butterfly valves, manually operated.

6. Main Steam

Design Conditions: 2750-psig pressure, 1005°F temperature

Pipe: 6 in. and larger: Hollow forged 2-1/4 chrome 1 molybdenum alloy steel, ASTM A335 Grade P22.
18-in. size 3.625-in. wall
14-in. size 2.625-in. wall
6-in. size 1.25-in. wall

2 in. and smaller: Seamless 2-1/4 chrome 1 molybdenum alloy steel, ASTM A335 Grade P22, double extra heavy.

Fittings:

2-1/2 in. and larger: 2-1/4 chrome 1 molybdenum alloy steel, ASTM A234 Grade WP22, butt-weld ends; wall thickness to match joining pipe.

2 in. and smaller: 6000-lb socket-weld forged 2-1/4 chrome 1 molybdenum alloy steel ASTM A182 Grade F22, ASA B16.11.

Joints:

2-1/2 in. and larger: Butt-welded with backing rings.

2 in. and smaller: Socket-welded.

Valves:

2-1/2 in. and larger: 2500-lb standard cast 2-1/4 chrome 1 molybdenum body and pressure seal bonnet, ASTM A237 Grade WC9, butt-weld ends, O.S. and Y, stainless steel trim solid wedge and plug disc, stellite seat and disc facing.

2 in. and smaller: 2500 lb standard forged 2-1/4 chrome 1 molybdenum body and welded on integral bonnet, ASTM A182 Grade F22, socket weld ends, O. S. and Y. stainless steel trim, solid wedge or plug disc, stellite seat and disc facing.

7. Hot Reheat Steam

Design Conditions:	600-psig pressure, 1005°F temperature
Pipe: 16 in. and larger:	Hollow forged 1-1/4 chrome 1/2 molybdenum alloy steel, ASTM A335 grade P11. 20-in. size 1.000-in. wall 16-in. size 0.875-in. wall.
14 in. and smaller:	Seamless 1-1/4 chrome 1/2 molybdenum alloy steel ASTM A335 Grade P11; 14-in. in size Schedule 100, 12-in. size and smaller Schedule 80.
Fittings:	
2-1/2 in. and larger:	1-1/4 chrome 1/2 molybdenum alloy steel, ASTM A234 Grade WP11, butt-weld ends, wall thickness to match joining pipe.
2 in. and smaller:	3000-lb socket-weld forged 1-1/4 chrome 1/2 molybdenum alloy steel, ASTM A182 Grade F11, ASA B16.11
Joints:	
2-1/2 in. and larger:	Butt-welded with backing rings.
2 in. and smaller:	Socket-welded.
Valves:	
2-1/2 in. and larger:	900-lb standard cast 1-1/4 chrome 1/2 molybdenum body and pressure seal bonnet, ASTM A217 Grade WC6, butt-weld ends, O. S. and Y. stainless steel trim, solid wedge and plug disc, stellite seat and disc facing.
2 in. and smaller:	1500-lb standard forged 1-1/4 chrome 1/2 molybdenum body and welded or integral bonnet, ASTM A182 Grade F11; socket-weld ends, O. S. and Y., stainless steel trim, solid wedge or plug disc, stellite seat and disc facing.

8. Cold Reheat Steam

Design Conditions:	600-psig pressure 640°F temperature
Pipe: All	Seamless carbon steel, ASTM A106 Grade B. 16-and 20-in. size Schedule 30. 14-in. size Schedule 30. 2-1/2-to 8-in. size Schedule 40. 2 in. and smaller Schedule 80.

Fittings:

2-1/2 in. and
Larger:

Carbon steel, ASTM A234 Grade WPA,
butt-weld ends, wall thickness to match
joining pipe.

2 in. and smaller:

3000-lb socket-wld forged carbon steel,
ASTM A105 Grade II, ASA B16.11.

Joints:

2-1/2 in. and larger:

Butt-welded with backing rings.

2 in. and smaller:

Socket-welded.

Valves:

2-1/2 in. and
larger:

600-lb standard cast carbon steel body and
bolted ring joint bonnet, ASTM A216 Grade
WCB, butt-weld ends, O. S. and Y., stainless
steel trim, solid wedge and plug disc,
stainless steel seat and disc facing.

2 in. and smaller:

600-lb standard forged carbon steel body
and welded or integral bonnet, ASA A105
Grade II; socket weld ends, O. S. and Y.,
stainless steel trim, solid wedge and plug
disc, stainless steel seat and disc facing.

9. Boiler Feed Discharge

Design Conditions:

3500-psig pressure, 500° F temperature

Pipe: All

Seamless carbon steel ASTM A106, Grade B,
10 in. size 1.313 in. wall; 8 in. and smaller
Schedule 160.

Fittings:

2-1/2 in. and
larger:

Carbon steel ASTM A234 Grade WPA, butt
weld ends, wall thickness to match joining
pipe.

2 in. and smaller:

6000-lb socket weld forged carbon steel,
ASTM A105 Grade II ASA B16.11

Joints:

2-1/2 in. and
larger:

Butt-welded with backing rings.

2 in. and smaller:

Socket-welded.

Valves:

2-1/2 in. and
larger:

2500-lb standard cast carbon steel body and pressure seal bonnet ASTM A216 Grade WCB, butt weld ends; O. S. and Y., stainless steel trim, solid wedge or plug disc, stellite seat and disc facing.

2 in. and smaller:

2500-lb standard forged carbon steel body and welded or integral bonnet, ASTM A105 Grade II, socket weld ends; O. S. and Y., stainless steel trim, solid wedge or plug disc, stellite seat and disc facing.

10. Extraction Steam

Design Conditions:

- a. 300-psig pressure, 500°F temperature H. P. heater 2.
- b. 150-psig pressure, 400°F temperature all others.

Pipe:

2-1/2 in. and
larger:

- a. Seamless carbon steel ASTM A106 Grade B 10-in. size, Schedule 60, 8 in. and smaller, Schedule 40.
- b. Seamless carbon steel ASTM A53 Grade B, 10-in. size and larger 0.375 in. wall, 8 in. and smaller Schedule 40.

2 in. and smaller:

Seamless carbon steel ASTM A106 Grade B, Schedule 80.

Fittings:

2-1/2 in. and
larger:

Carbon steel ASTM A234 Grade WPA, butt-weld ends; wall thickness to match joining pipe.

2 in. and smaller:

3000-lb socket-weld forged carbon steel, ASTM A105 Grade II ASA B16.11.

Joints:

2-1/2 in. and
larger:

Butt-welded with backing rings.

2 in. and smaller:

Socket-welded.

Valves:

2-1/2 in. and
larger:

- a. 300 lb

- b. 150-lb cast carbon steel body and bolted bonnet, ASTM A216 Grade WCB, butt-weld ends; O. S. and Y., stainless steel trim, solid wedge or plug disc, stainless steel seat and disc facing.

2 in. and smaller:

600-lb standard forged carbon steel body and welded or integral bonnet, ASTM A105 Grade II, socket-weld ends; O. S. and Y., stainless steel trim solid wedge or plug disc, stainless steel seat and disc facing.

11. Condensate and Boiler Feed Suction

Design Conditions:

- a. 300-psig pressure, 300°F temperature for condensate pump discharge to L.P. heater 7.
- b. 150-psig pressure 400°F temperature, all others.

Pipe:

2-1/2 in. and
larger:

Seamless carbon steel ASTM A53 Grade B, 12-in. size and larger Schedule 30, 10-in. and smaller Schedule 40.

2 in. and smaller:

Seamless carbon steel ASTM A106 Grade B Schedule 80.

Fittings:

2-1/2 in. and
larger:

Carbon steel ASTM A234 Grade WPA, butt-weld ends; wall thickness to match joining pipe.

2 in. and smaller:

3000-lb socket-weld forged carbon steel, ASTM A105 Grade II, ASA B16.11

Joints:

2-1/2 in. and
larger:

Butt-welded with backing rings.

2 in. and smaller:

Socket-welded.

Valves:

2-1/2 in. and
larger:

- a. 300 lb
- b. 150-lb cast carbon steel body and bolted bonnet, ASTM A216 Grade WCB, butt-weld ends; O. S. and Y., stainless steel trim, solid wedge or plug disc, stainless steel seat and disc facing.

2 in. and smaller: 600-lb standard forged carbon steel body and welded or integral bonnet, ASTM A105 Grade II, socket-weld ends; O. S. and Y. stainless steel trim, solid wedge or plug disc, stainless steel seat and disc facing.

B. THERMAL INSULATION

The thermal insulation material for high-temperature (500°F and over) systems is hydrous calcium silicate or diatomaceous silica combined with asbestos.

The low-temperature (under 500°F) insulation is 85% magnesia, fibreglass or hydrous calcium silicate.

Personnel protection insulation on piping is installed where required on piping from the floor up to height of 8 ft 0 in. or up to 8 ft 0 in. directly above any platform or walkway, and to a distance of 1 ft 0 in. beyond any point where personnel can normally reach. This hazard is interpreted to exist within 3 ft 6 in. of any walkway or platform. The material to be used for covering piping to prevent sweating is a sectional molded covering. The antisweat insulation will be suitable to prevent sweating with a metal temperature of 50°F, and ambient air temperature of 90°F and a relative humidity of 80%. Cork-filled mastic suitable for trowel or spray application is used to prevent sweating on equipment and circulating water piping within the building.

XI. OPERATION AND MAINTENANCE

Operating and maintenance procedures are based on experience obtained from the operation of the Sodium Reactor Experiment and the Hallam Nuclear Power Facility. This experience has demonstrated expected operating characteristics and indicated the type and number of personnel required to operate the proposed prototype power plant.

A. OPERATING STAFF

A suggested operations organization for the plant is shown in Figure 12. The total operating and maintenance staff for a single-unit station is estimated to be 68. Later expansion would require proportionately fewer additional personnel. With several power plants in the same area, a common labor pool would also reduce the permanent maintenance personnel of the plant.

Resident engineer requirements are also affected by the proximity and availability of engineers from general offices for consultation. In general, the following technical personnel are believed to be required.

- 1) Reactor Engineer: to maintain surveillance of reactor data, determine and report reactor reactivity coefficients, and make recommendations to improve performance and fuel management.
- 2) Mechanical-electrical Engineer: to keep a history of all equipment in the plant and make recommendations to improve performance.
- 3) Health Physicist: to maintain surveillance of plant safety and procedures involving sodium and radioactive equipment.
- 4) Instrumentation Engineer: to maintain surveillance of all plant instrumentation, analyze equipment malfunctions, define instrumentation requirements for heat-balance tests and for obtaining special data, and make recommendations to improve performance.
- 5) Efficiency Engineer: to maintain surveillance of all heat exchange equipment so as to detect degradation of plant performance, to maintain over-all plant performance, and to keep overall plant performance records.

Figure 12. Operations Organization

- 6) Chemist: to maintain surveillance of all plant fluids to ensure proper purity and treatment of boiler water, cooling water, sodium, helium, nitrogen, and to recommend proper methods of washing reactor components.

The shift supervisor is responsible for the safe and proper operation of the plant. Normal operation of the plant requires a control operator in constant attendance at the reactor and turbine controls. An assistant control operator is required to provide surveillance of auxiliary indicators and recorders, and to log the data required to maintain adequate records of plant performance.

Roving attendants are provided for necessary data-taking and surveillance of equipment and instrumentation of the turbine plant, reactor plant, and miscellaneous outside equipment. An instrument technician is available on each shift to perform adjustment and preventive maintenance on the instrumentation systems.

B. PLANT OPERATION

1. Normal Operation

The automatic control system, together with the inherent stable characteristics of the sodium graphite reactor, results in a plant which is easily operated as a load-following or base-load plant at either partial-capacity or full-power operation as indicated by the following recommendations.

- a) The control system provides variable flow so that system temperatures will remain approximately constant as the plant load is varied.
- b) Reactor system response is more than adequate to supply energy to the turbine at any foreseeable load rate change.
- c) Automatic operation is possible over a load range from 20 to 110% of rated power.
- d) The plant is controlled, during both steady-state and normal load changes, by the fully automatic control system.
- e) The operators maintain continuous surveillance of important plant variables to ascertain the proper operation of the controls and log the data required for plant records.

- f) Deviation alarms on important control points require the operators to make manual control of the affected system until the deviation can be corrected, the controller realigned, and the system returned to fully automatic control.

2. Cold Startup

A cold startup is defined as any startup with less than 50°F temperature differential across the reactor. In the extreme case, all heat-transfer loops would be drained and cold. For such a startup the following steps would be followed.

- a) Preheat the sodium piping and equipment to 350°F. The sodium piping and intermediate heat exchangers are preheated with electrical heaters. The steam generator and reheater are heated by circulating boiler feedwater, using steam from the flash-tank and the auxiliary boiler in the feedwater heater. The heating rate is controlled to a nominal 50°F/hr temperature increase.
- b) Fill the sodium loops. Normally the secondary loops are filled first to assure even heat distribution in the intermediate heat exchangers. Once the systems are filled, sodium is circulated and maintained at a minimum temperature of 350°F, using steam from the auxiliary boiler to prevent the steam generators from becoming a heat sink.
- c) Take the reactor critical, check reactivity, increase power to level required to raise system temperature at the design rate. This will require about 1% design power.
- d) Raise isothermal temperature of the entire system to design reactor inlet temperature. Temperature differential will have to be established to generate steam for the flash-tank.
- e) Clean up the feedwater system. Circulate feedwater through the low temperature modules and demineralizers. The feedwater temperature is increased by reactor power as the sodium temperature increases.
- f) Dump steam to the steam-dump system until steam reaches a condition satisfactory for rolling and synchronizing the turbine.

- g) Balance turbine load and steam generation, simultaneously close off steam-dump system and prepare it for standby service.
- h) Establish design temperatures across the reactor by adjusting reactor power to continue 50°F/hr temperature increase.
- i) Continue to raise reactor power and turbine load on semi-automatic control up to approximately 20% design power.
- j) Place plant in fully automatic control and supply demand power.

3. Hot Startup

A hot startup is a startup after a short shutdown when more than a 50°F temperature differential exists across the reactor. The heat transfer loops are filled and sodium circulation is controlled by the thermal convection control system. Here are the major steps to be followed.

- a) Take reactor critical, check reactivity, increase power to level required to raise system temperatures at the design rate.
- b) As soon as temperatures begin to increase, the convection flow control system will open the throttling valves.
- c) As the valves begin to open, slowly start the sodium pumps and establish forced sodium circulation.
- d) Adjust reactor power and pump speeds to continue system heatup at the design rate.
- e) Once valves are open, pick up the appropriate step in the cold startup procedure and continue to desired power level. It may be necessary to desuperheat main and reheat steam temperatures to match turbine metal temperatures.

4. Normal Shutdown

A normal shutdown is defined as a scheduled outage. Major steps that occur are as follows:

- a) Shut down the turbine following conventional patterns of reducing to minimum load and tripping the turbine,
- b) Alternatively, if desired, reduce turbine load, steam temperature, and sodium temperature to minimum values, and then trip.

Steam generator operation must continue, together with the warmup and steam-dump system to provide a heat sink for decay heat removal from the reactor. The design of these systems makes them independent of the conventional plant equipment so that any required maintenance on the conventional system equipment can be carried out with no interference from the decay heat removal operation. The decay heat removal and the Sodium²⁴ decay take place concurrently so there is no lost time for maintenance of primary sodium equipment because of decay heat removal.

5. Reactor Scram

The following steps take place after a reactor scram.

- a) The turbine is isolated.
- b) The outlet temperature of steam is reduced by desuperheating to 750°F before the steam is admitted to the atmospheric boiler.
- c) Steam flow through the reheat and high-temperature modules is controlled from the flash-tank to remove excess heat. Flash-tank steam continues to be supplied to the module feedwater heater, and the turbine-driven boiler feedwater pump is automatically started. The capacity of this pump is 10% of rated flow, which is adequate for decay heat removal.
- d) In the reactor heat-transfer loops, sodium flow is controlled by the convection flow control system.
- e) Depending on the reason for the scram and the plans for future operation, the plant may be cooled down following the normal shutdown procedures, or the reactor may be restarted in accordance with the hot startup procedure.
- f) If the plant is to be shut down, the system is cooled using the decay heat removal system.
- g) As pressure in the flash-tank decays, and the sodium and feedwater temperatures start dropping, the turbine-driven pump can no longer function. To compensate for the loss of power, two small electric pumps are provided. These pumps assure water flow for heat dissipation when the auxiliary steam boiler cannot be operated.

6. Turbine Trip

In the event of a turbine trip, the following steps take place.

- a) The steam dump system valves open automatically to dissipate part of the steam; the pump turbine starts with automatic shutdown of the main pumps, and the module feedwater heater continues in operation.
- b) The reactor control system automatically lowers the reactor power level at the maximum controllable rate.
- c) If steam pressure increases, the power relief and, if necessary, the safety valves open to discharge the excess steam to the atmosphere.
- d) The reactor power level continues to drop until it matches the capability of the steam dump system.
- e) At this point the turbine may be restarted following standard startup procedures; or, if necessary, the plant can be shut down following the normal shutdown procedures.

7. Emergency Conditions

In the event of serious off-normal conditions which cannot be handled by the control system:

- a) The plant protective system detects the condition, actuates alarms, and, when necessary automatically reduces reactor power level and coolant flow; and
- b) Operator action is required to determine the cause of the alarm and take appropriate corrective action in accordance with standard corrective action procedures.

C. MAINTENANCE

1. Conventional Equipment

Most of the equipment in the plant can be described as "conventional", i.e., standard equipment similar to that in many power or chemical plants. Maintenance of this equipment is in accordance with standard manufacturer's recommendations and requires only the usual careful attention given any plant component.

2. Sodium System Equipment

Equipment from the sodium systems is often similar to equivalent components in other parts of the plant, but maintenance requires additional effort since the equipment may contain or be covered with sodium. Working with sodium requires special training of maintenance personnel, but when handled according to established procedures, sodium presents no undue hazards. Usually, before the required maintenance can be performed, the sodium must be cleaned from the equipment. This can be easily accomplished by low-pressure saturated steam spray, or by soaking in n-butyl alcohol. Equipment is provided in the plant to expedite such cleaning.

3. Reactor and Primary System Equipment

Maintenance of reactor and primary system equipment is complicated by the possibility of radioactive contamination. All items removed from these areas must be carefully checked for such contamination before being handled or moved. Such monitoring minimizes personnel danger and avoids spreading radioactive contamination throughout the plant. Often, special clothing is used to further control the spread of radioactive contamination. Special procedures are prepared to cover the handling of reactor and primary system components and, if followed properly, will ensure safe, efficient handling of this equipment.

Since the primary sodium becomes radioactive, maintenance on primary system components will require loop shutdown of approximately 10 days for radioactivity decay. Access to shielded areas will require the reactor to be off the line. All primary sodium systems are in cells which operate in a nitrogen atmosphere. It is necessary to purge the cell and ascertain low oxygen content and radioactivity levels before entry is permitted. Radioactive material-handling precautions are required until the radioactivity level of the equipment is determined.

There are five items in the primary system which might require maintenance; pumps, heat-exchangers, traps, valves, and instrumentation. The pumps are designed for removal for maintenance and inspection operations without requiring cell entry. Heat-exchangers are not expected to require frequent maintenance, and other items are engineered to require minimum cell entry. Control elements, which are electrically sensed, are provided with spare leads and elements so that operation can continue without requiring a shutdown and decay time to make repairs.

Maintenance of all potentially radioactive equipment is done under the cognizance of the Health and Safety staff which establish the safety precautions necessary to assure that exposure and dose rates are within prescribed limits.

Maintenance of components having a high level of radioactivity which does not allow contact work is done in the maintenance cell. The cell is equipped with remote-handling gear which allows inspection, removal, and replacement of in-core components which can be transported by the fuel-handling machine. Provisions are also available within the cell for the packaging of equipment for disposal.

A component wash-cell is provided to wash the sodium pumps and other sodium-contaminated equipment, including such items as control rod thimbles, should a need for washing this equipment ever arise.

The reactor is designed so that normal maintenance operations can be performed by operations and maintenance personnel. Biological shielding is provided around all components so that maximum estimated radiation level in full-time occupancy areas will not exceed 0.75 mrem/hr. Access to these areas is possible at all time during normal operation of the facility.

All fuel-handling operations other than reactor fueling can be done independently of reactor operation. These operations are accomplished directly by use of the fuel-handling machine and the maintenance cell.

The calandria is designed to permit maintenance of the process channels. The channels consist of the process tube, bellows, and moderator graphite. The process tube is fitted at both ends with joints which can be brazed and leak-tested with tooling designed for this purpose. Graphite in the calandria vessel is of two general forms: round logs and filler logs. Special tooling will be required for the installation and removal of both forms of graphite. In the event of abnormal maintenance on the calandria, the fuel-handling machine can be used for insertion and removal of all components associated with this maintenance.

4. Preventive Maintenance Program

To minimize failures and extend the life of plant equipment, a preventive maintenance program should be conscientiously followed. This program should follow the recommendations of the equipment vendors and take advantage of the experience gained at the HNPF regarding expected service life and maintenance frequency of sodium and reactor components. The preventive maintenance program will be conducted by the plant maintenance personnel with assistance, as required, by vendor servicemen.

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XII. REACTOR SAFEGUARDS

This section considers the safeguards aspects of plant site selection. The inherent and design safety features of SGR's and their mitigating effect on the results of various accidents are discussed. The application of these results to the AEC reference site is considered. As shown below, the SGR prototype described in this report can be located at close-in sites and meet the requirements of AEC regulations in "Reactor Siting Criteria," Title 10 of the Code of Federal Regulation, Part 100, (10 CFR 100).

A. SAFETY FEATURES

The safety features of an SGR system which permit installation of a plant at any typical electrical utility site fall into three categories: (1) inherent safety features resulting primarily from the chemical and physical properties of sodium; (2) general design features required to accommodate these properties and necessary to meet standard plant design objectives; and (3) engineered safeguards or design features expressly required for safety purposes which can be varied to meet the necessary requirements of a particular site.

1. Inherent Features

The inherent safety features of sodium-cooled reactors are primarily due to the chemical and physical properties of sodium. The more significant properties of sodium which affect the safety of sodium graphite system are shown in Table 11.

TABLE 11
SODIUM PROPERTIES AFFECTING SAFETY OF SODIUM
GRAPHITE REACTORS

Vapor pressure	low
Boiling point	high
Heat capacity	high
Thermal conductivity	very high
Chemical affinity	high
Fire potential	pyrophoric
Radioactivity	high
Decomposition	none
Fuel-sodium reaction	none

In the design of a sodium graphite reactor system, full advantage is taken of those properties which minimize the severity of accidents; also, thorough precautionary measures are taken with those properties which could have a deleterious effect in reactor accidents.

a. Vapor Pressure

The low vapor pressure of sodium at reactor operating temperatures permits reactor system operation at very low pressures, essentially atmospheric. If there were a loss of primary system integrity, there would be no immediate release of large quantities of stored energy resulting in the vaporization of the coolant and pressurization of the secondary enclosure.

b. Boiling Point and Heat Capacity

The operating temperature of the reactor system is 400 to 600°F below the boiling point of sodium at atmospheric pressure. Thus, the sodium has the capacity for absorbing a large quantity of sensible heat without boiling. This provides a margin of safety to absorb abnormal energy releases without damage to the fuel or over-pressurization of the reactor system by boiling.

c. High Burnout Heat Flux

Another important thermal characteristic of sodium is its unusually high burnout heat flux. This permits operation at relatively high thermal fluxes which still have large safety margins relative to burnout. Because of the high heat flux permitted, sodium-cooled reactors have high specific power (kw/kg uranium) and consequently lower working capital requirements than that of any other proposed reactor.

d. Compatibility with Various Materials

Even at high temperatures, sodium is fully compatible with ordinary structural materials such as ferritic and austenitic steels. Sodium has demonstrated chemical compatibility with such materials as uranium, uranium alloys, uranium carbide, uranium oxide, graphite, and zircaloy. The absence of a significant chemical reaction between sodium and the reactor fuels used in sodium graphite reactors eliminates the possibility of chemical energy contributing to an accident following a loss of cladding integrity. There are no sources of chemical energy within the primary system which could result in explosions or shockwaves and subsequent damage to that system.

e. Chemical Affinity

The chemical properties of sodium result in highly efficient trapping of all fission products with the exception of the noble gases. Since the coolant does not flash and because of its extremely great chemical affinity for iodine, sodium is an effective trapping agent which prevents the release of iodine to the environment. This is particularly important because of the role that iodine plays in reactor accident evaluation. Regulation 10CFR 100 defines two specific radiological effects on individuals which must be determined in order to analyze the relative severity of an accident. The first is the radiation dose to the whole body, and the second is the radiation dose to the thyroid from iodine exposure. If any appreciable quantity of radioactive iodine is released, the second criteria tends to dominate the severity of the accident. With iodine unavailable for release, even assuming a conventional building as containment, required exclusion distances are greatly reduced.

f. Induced Radioactivity

Sodium becomes radioactive when passing through the reactor, so the primary coolant system must be surrounded with a biological radiation shield. To eliminate the possibility of radioactivity being introduced into the steam system, a nonradioactive secondary sodium coolant system is employed. The secondary system requires no shielding.

g. Fire Potential

Sodium reacts chemically with both air and water, so care must be exercised to provide a well-sealed system. Wherever radioactive sodium is present, an inert gas environment is provided for all storage and handling. While it is pyrophoric in the presence of oxygen, there is substantial experimental evidence which indicates that a sodium fire is self-extinguishing. During the combustion of sodium, an oxide crust forms on the surface of the burning sodium pool which extinguishes the fire in less than 20 hr. Experiments show that self-extinguishing occurs when the thickness of the oxide crust exceeds 8 in. This results in a limited quantity of radioactive sodium oxide ash. The same experiments show that fallout, condensation, and deposition of the ash result in appreciable reduction in the amount of radioactivity which would otherwise be available for dispersion to the environment. Furthermore, there is evidence to indicate that certain fission product isotopes (notably iodine) tend to remain in the burned residue of sodium rather than be released with the escaping fumes.

The significant area for possible contact with water is in the steam generator. The system is designed to minimize the probability and the effect of steam generator tube rupture.

2. General Design

The SGR is designed to take full advantage of the safety characteristics of sodium-graphite systems.

a. Graphite Heat Capacity

The large heat capacity of graphite in the reactor provides additional capacity for decay heat removal from fuel elements without melting of the fuel or pressurizing the system.

b. Embrittlement

Due to the high temperatures at which an SGR operates, high steady-state thermal stresses occur. These stresses are accommodated by the use of stainless steel in the system. The likelihood of brittle failure of the reactor vessel is reduced by the use of stainless steel since it is resistant to changes in the nil-ductility temperature transition point by high energy neutron bombardment.

c. Nuclear Coefficients

The main mechanism for the prompt shutdown (and a major contributor to normal control and stability) of reactor system is the prompt temperature coefficient of reactivity. The major component of this coefficient in sodium graphite reactors is strongly negative due to the Doppler broadening of resonances in low-enrichment fuels. Furthermore, due to the high thermal conductivity of uranium carbide, the average fuel temperature is relatively low. Thus a large amount of prompt (less than 1 sec) negative reactivity is available to assist in limiting the effect of a nuclear excursion. The major positive reactivity coefficient in sodium graphite reactors arises from the temperature coefficient of graphite and the void coefficient of sodium. The time constants for these positive reactivity coefficients are sufficiently long (minutes) to permit control by protective systems.

d. Core Coolant Draining

Both inlet and outlet lines of the reactor vessel are built with guard piping to an elevation above the reactor outlet to prevent uncovering of the core

by coolant draining from the vessel should a leak occur in the primary piping. Core exposure by gravity draining is also prevented by a cavity liner around the reactor vessel to contain the sodium and maintain its level above the core in the event of gross leakage from the reactor vessel. In addition, loss of coolant by pressure transfer through a primary pipe rupture is prevented by guard piping around the inlet piping from the reactor vessel to the first valve.

e. Secondary Containment

An outer vessel and a cavity liner provides secondary containment for the sodium in the event of a primary vessel failure. The remainder of the primary system is located in gas-tight concrete cells filled with an inert gas to prevent the possibility of sustained sodium combustion in the event of primary system failure. Thus, in the unlikely event of a primary system failure, there is no pressure rise due to flashing of the coolant or oxidation of the sodium (without violation of the secondary containment vaults).

f. Secondary Sodium System

The chemical reaction of water with hot sodium results in generation of hydrogen. Therefore, water is excluded from possible contact with primary system sodium. Heat is removed from the primary sodium by a secondary sodium in an intermediate heat exchanger. Since the secondary sodium system does not become radioactive, it does not require shielding or containment. Thus, the heat exchanger for the secondary sodium system, the steam generator, is located outside of the reactor building.

3. Engineered Safeguards

The engineered safeguards of SGR systems include design variables which may be required for safety at a particular site. The large number of inherent safeguards in sodium graphite reactor systems permits greater flexibility in selection of a site. This flexibility allows the use of more sites or the better use of existing sites to meet the power requirements of the service area.

a. Leak Rates

The permissible leak rate from containment is dependent upon the plant site and the use of multiple sequential containments. It is necessary to compute the maximum pressure rise within a containment zone in order to assure that the leak rate criteria can be met while containment zone integrity is maintained.

(1) Cells

Normally, the leak rate requirement of cells is more heavily dependent upon conservation of the inert gas than the hazards arising from accident evaluation. Taking into account the pressure-time functions arising from sodium leaks (including normal energy releases due to limited oxygen availability), SGR primary containment cells are designed with leak rates as low as 1% of the contained volume in 24-hr-per-psi pressure differential.

(2) Building

Sodium graphite reactors do not require pressure containment structures. In many cases, standard industrial buildings with leak rates in excess of several thousand percent of the contained volume per day can be used in siting factors are favorable and controlled release through the ventilation system is permitted. Carefully designed and constructed buildings of conventional materials can be built with leakage as low as 1% of the contained volume per day at low pressure (1/2 psi) for a modest cost increase over conventional construction. Hermetically sealed structures for reactor buildings have been constructed with the capability of leaking less than 0.1% per day at a design pressure of 30 psi.

b. Air Cleaning

Chemical and mechanical air cleaning devices can be used to mitigate the effect of a radiological release following an accident. Many of these devices have cleaning efficiencies in excess of 99%. An internal cleanup system could be installed in the reactor building to clean the atmosphere of secondary cells or vaults.

The disposal of gaseous waste material is regulated by the meteorology and ecology of the plant's location. Holdup (time delay for decay before release) is routinely used in sodium graphite reactor systems. The use of particulate removal systems or halogen traps is effective in removing radioactive contaminants.

c. Stack

A ventilation stack is effective in reducing exposures by many orders of magnitude, depending upon the height and distances from the stack. The dilutions available range as high as 1/1000. The dilutions required depend primarily upon meteorology, population distribution, and pretreatment of airborne effluent.

d. Liquid Effluents

Liquid radioactive waste from sodium graphite reactors arises primarily from washing of primary system components under maintenance, and radiochemical laboratory operations. The quantities involved do not normally constitute either a safety problem or a disposal problem.

e. Shielding

A considerable reduction in direct dose can be realized by employing concrete shielding for the reactor building. Utilization of such shielding may be desirable or necessary at a close-in site. However, a totally unshielded building is possible where large distances exist to large population centers. Because of cost, it is desirable to provide the minimum necessary shielding for any site.

B. RADIOLOGICAL HAZARDS ANALYSIS

1. Introduction

A preliminary radiological hazards analysis was performed for the SGR prototype located at the reference site. In order to appropriately evaluate a particular reactor-site combination, information must be developed to assess the potential hazard to off-site personnel. Three specific distances from a proposed reactor location are utilized in determining the suitability of a particular site. These distances define the boundary of the following area which are specified as follows: (a) exclusion area — the area immediately surrounding the reactor in which the reactor operator has complete control of all personnel and activity; (b) low population zone — the area surrounding the exclusion area which can be evacuated in an emergency; and (c) population center — the required distance to a population center must not include any portion of a population center of 25,000 or more inhabitants.

The 10CFR 100 regulations recommend an "exclusion area" of adequate size so that a release of radioactive material would result in a radiation dose of less than 25 rem to the whole body or 300 rem to the thyroid during the first 2 hr; a "low population zone" boundary distance that would result in these same dose limits during the entire duration of the accident; and a "population center distance" of 1-1/3 times the low population zone distance. Obviously, the safer the reactor, the shorter these distances may be.

The hypothetical AEC reference site is located on the east bank of a river, 35 mi north of a large city of population 250,000. Variation in population with distance from the site boundary is shown in Table 12. The nearest residence to the plant is 3/8 mi east of the site boundary. The shortest distance from the reactor to the site boundary is 0.4 mi.

TABLE 12
SITE AREA POPULATION

Distance from Site (mi)	Population
0.25	0
0.5	60
1.0	200
5.0	2,700
10.0	8,000
20.0	40,000

There are five industrial manufacturing plants within 15 mi of the site. These are small plants employing less than 100 people each. Closely populated areas are found only in the centers of the small towns so the total land area used for housing is small. The remaining land, including that across the river, is used as forest or cultivated crop land, except for railroads and highways.

2. Significant Accidents

A variety of accidents may be postulated for any given plant. However, those which are meaningful for site selection are those which have the potential of producing a significant hazard to the general public. Based on the experience gained by the design, construction, operation, and safeguards analyses of the Sodium Reactor Experiment and the Hallam Nuclear Power Facility, three selected reference accidents have been evolved which represent the upper limit for public hazards. These are (a) core meltdown, (b) fuel-handling accidents, and (c) sodium fires. The probability of these accidents occurring is extremely low, and requires multifailures of the reactor system and protection system components.

In the accident analysis, the meteorological conditions were assumed to be described by Pasquill's Type F parameters. Type F parameters represent atmospheric conditions that would result in the maximum downwind concentrations at population center distances and, subsequently, the maximum radiation dosages

to the greatest number of off-site personnel. For stack releases, other meteorological conditions, e.g., Pasquill's Type A or C, could result in higher concentrations around smaller population groups at close-in distances, but lower concentrations than the Type F conditions at large distances. For those cases where the close-in concentrations result in significant doses, Type A and C meteorological conditions are analyzed.

The AISITE* computer code was used for calculation of the thyroid dose resulting from the accidents. Whole body cloud doses were calculated with the Thunderhead† code. Direct radiation from activity retained in the reactor building was calculated with consideration of the shielding provided by the 6-in.-thick concrete building walls.

a. Core Meltdown

Core overheating caused by an uncontrolled power excursion would be checked (but not necessarily stopped) by the negative fuel temperature coefficient of reactivity. The ultimate shutdown mechanism is the reduction of the core to a subcritical mass by melting and relocation of fuel. This would terminate an uncontrolled excursion. Such an excursion might result from continuous control rod withdrawal, coupled with failure of normal protective system scram action, although this sequence of events is very unlikely. Reactivity calculations indicate that melting and relocation of one-third of the fuel would be sufficient to shut down the reactor even with complete rod withdrawal.

In the core meltdown accident it is assumed that one-third of the core fission product inventory is released to the sodium coolant. Noble gases will be subsequently released to the core cover-gas. The bulk shielding of the primary system will prevent the direct radiation dose from the system from being of consequence even to operating personnel. The reactor cover-gas is normally fully contained, with less than detectable leakage. Also, it can be purged and vented to the vent decay tanks for safe storage. However, in this analysis, it is assumed that the cover-gas leaks to the high-bay area at a rate of 20 volume %/day. This is the estimated leakage rate around a fuel shield plug, the sealing quadrings of which have been removed, based upon HNPF studies. Other assumptions are listed in Table 13.

*R. A. Blaine, "AISITE: A Reactor Siting Code," AMTD-203 (May 1963).

†G. P. Kenfield and W. B. Sayer, "Thunderhead - An External Cloud Dose Code," NAA-SR-MEMO-8952 (unpublished).

TABLE 13

ASSUMPTIONS USED TO CALCULATE ENVIRONMENTAL
DOSES FROM CORE MELTDOWN ACCIDENT

Reactor operating time; full power (days)	300
Reactor power (Mwt)	480
Cooling	0
Fraction of the fuel melted	1/3
Building leak rate	145 volume %/day at 1/4-in. H ₂ O
Cover-gas leak rate	20 volume %/day
Stack height (meters)	60
(ft)	197
Noble gas release fraction	
from sodium coolant	1.0
from building to filters	1.0
through filters	1.0
Halogen release fraction	
from sodium coolant	2×10^{-4}
from building to filters	0.5
through filters	0.05

In the event of significant cover-gas leakage following core meltdown, automatic action triggered by a radiation detector in the exhaust duct will shut off the main building fans and inlet air and start the emergency fan. This will maintain the building pressure at -1/4-in. H₂O, while discharging in-leakage air through the high-efficiency and halogen filters. Noble gases and a small amount of halogens will be discharged from the stack with the effluent.

The 30-day whole-body dose as a function of distance for this accident, assuming Type F meteorology* which, for the reference site, gives the highest dose to the greatest number of people, is presented in Figure 13. The excellent retention of halogens by the sodium coolant reduces the thyroid dose to an insignificant value. The only significant dose, from the standpoint of site criteria, is the direct dose due to fission products in the high bay. This dose requires the low-population zone boundary to be 0.07 mi from the reactor, which is less than the distance found for the fuel-handling accident; and, in addition, is well within a reasonable site boundary. The maximum off-site cloud dose is ~4.7 rem from noble gases.

*F. Pasquill, Atmospheric Diffusion, Van Nostrand, New York (1962).

Figure 13. Core Meltdown Accident

b. Fuel Handling

The fuel-handling accident involving an entire fuel element is not considered credible in this plant, but it is analyzed as a classic accident which can be used for facility comparisons. It is assumed that a complete, irradiated fuel element is dropped on the floor of the reactor building during fuel-handling activities and is consumed by oxidation, thus releasing fission products into the building. The results of this accident can be mitigated by the extent to which engineered safeguards are employed. For example, the design of the reactor building with respect to the leak rate and the use of shielding can be varied. In addition, the gaseous effluent disposal system (particularly the extend of filtration before release to the atmosphere) and stack height are effective in controlling the results of this type of accident.

To determine the upper limit of the consequences from this accident, complete oxidation of the element in 15 min is assumed. It is also assumed that all the heat from the UC oxidation is released directly to the building atmosphere. Assuming that the original building temperature is 70°F, the temperature after 15 min is 92°F for the approximately 1.1-million-ft³ building, and about 50,000ft³ of air must be released to avoid pressurizing the building.

In order to determine the magnitude of the potential environmental effects of this accident, particular design features of the plant must be considered. Normal ventilation exhaust air at 200,000 cfm bypasses the filter system. A radiation detector in the exhaust duct would automatically shut off exhaust and inlet fans, and activate the emergency vent system in event of a radioactive release. The emergency ventilation system would remove building air at a rate of 3000 cfm during the 15 min of fuel oxidation, and then release building in-leakage air at the design leak rate of 1150 cfm at 1/4-in. water static pressure (145%/day) for the rest of the exposure period (a few days). With this system operative, all of the significant released activity would be discharged via the stack and filter system. The filter system includes activated charcoal halogen filters and high efficiency particulate filters. Other assumptions are given in Table 14. The resultant 30-day dose as a function of distance from the reactor is presented for the three types of meteorological conditions in Figure 14. These results show the 30-day max thyroid dose to be only 5.5 rem and thus not a limiting siting factor. The 30-day whole-body dose from direct radiation is 25 rem at 0.21 mi; therefore, this is the low-population zone distance for this accident, and might well lie within an ordinary site boundary.

TABLE 14

ASSUMPTIONS USED TO CALCULATE ENVIRONMENTAL
DOSES FROM FUEL HANDLING ACCIDENT

Reactor operating time; full power (days)	300
Peak element power (Mwt)	4.0
Cooling time (hr)	10
Oxidation time (min)	15
Fraction oxidized	1.0
Building design leak rate	145 volume %/day at 1/4-in. H ₂ O
Stack height (meters)	60
(ft)	197

Fission Products	Release Fraction to Building	Release Fraction from Building Atmosphere	Filter Transmission
Halogens	0.6*	0.5†	0.5§
Noble gases	1.0	1.0	1.0

*J. J. DiNunno et al, "Calculations of Distance Factors for Power and Test Reactor Sites," TID-14844, Division of Licensing and Regulation, USAEC, Washington, D. C. (March 23, 1962); and "Proceedings of the Uranium Carbide Meeting Held at Oak Ridge National Laboratory, Oak Ridge, Tennessee, December 1, 1960," USAEC, TID-7603

†J. J. DiNunno, op cit; and A. C. Chamberlin, "Relation Between Measurement of Deposited Activity After the Windscale Accident of October 1957," AERE HP/R-2606, UKAEA (September 1958)

§Manufacturers rated filter efficiencies for Halogen filters are greater than 99%. A more conservative 95% has been used here to account for possible, but unproven, decrease in efficiency with use.

c. Sodium Fires

Two major sodium fires were considered for comparison with the fuel handling accident in the analysis of the site. These have been labeled the "radioactive" and the "nonradioactive" sodium fires.

(1) Radioactive

Burning of sodium requires exposure to an atmosphere containing more than 10% oxygen. The radiological hazard results from the airborne release of fission products and the radioactive sodium-24 isotope, which are

Figure 14. Fuel Handling Accident

significant only in the primary sodium. During normal operation the fission product activity in the primary sodium will be insignificant relative to the sodium activity. Because all primary sodium equipment is contained in cells with inert-gas atmospheres, the only place that a fire can occur with sodium-24 present is in the sodium service vault, where cold traps are removed while the reactor is in operation. However, a minimum of two valves are secured between the main primary sodium line and the equipment, along with at least 30 ft of frozen sodium in the line, prior to equipment removal. The vault is constructed so that, in the unlikely event of a sodium leak and subsequent fire, it is sealed in 5 min and nitrogen coolant introduced to assist in smothering the fire and removing the heat. Thus, the worst radiological consequence of a sodium fire is a 5-min release of combustion production from the vault.

In this accident, since a 5-min delay in closing the sodium service vault would be maximum under anticipated conditions, it is the effect during this burning time that is analyzed. The maximum measured sodium burning rate of 2.5 lb/hr-ft² is used as the burning rate for the 5 min; in which case, the heat due to burning would raise the temperature of the building air about 7.7°F, with a resultant volumetric expansion of building air one-third as great as that due to the fuel handling accident. The data and/or assumptions used to calculate the required site boundary for this accident are presented in Table 15.

The results of the calculation for this case show that no special precautions are required and that the ventilation system can be permitted to function at normal 200,000-cfm release to stack during the entire course of the accident. (Although the radiation detector in the ventilation system would automatically route radioactive building exhaust air through the emergency system, operator action would restart the main ventilation fans in event of a sodium fire.) Figure 15 shows the 30-day whole-body dose vs distance for this accident for Type F meteorology. The Types A and C meteorology and thyroid dose are not significant.

The accident was further analyzed to study the pressure buildup in the vault subsequent to closing the gas-lock. If 10,000 lb of 900°F sodium is assumed to be on the floor of the vault, it is found that a maximum cell pressure of 3.3 psig is reached about 1 hr after the accident originates.

Cell pressure relief is normally through the radioactive vent system, with relief valves opening at approximately 2 psig. This system can vent

TABLE 15

ASSUMPTIONS USED TO CALCULATE ENVIRONMENTAL
DOSES FROM RADIOACTIVE SODIUM FIRES

Reactor operating time (days) (Na-24 reaches saturation in 4 days)	300
Reactor (Mwt)	480
Cooling time	0
Fraction of fission products in coolant	10^{-4}
Coolant burning time (min)	5
Coolant burning rate (initial burning rate for 5 min)	2.5 lb/hr-ft ²
Specific activity of Na (curies/gm)	0.25
Surface area (ft ²)	620
Total coolant burned (lb) (Total sodium inventory 5×10^5 lb)	130
Stack height (meters) (ft)	60 197
Stack flow rate (cfm)	200,000
No filters	

Release of Significant Fission Products Plus Sodium	Release Fraction to Building	Release Fraction to Stack
halogens	1.0	0.5
noble gases	1.0	1.0
sodium	0.5	0.5

Figure 15. Radioactive Sodium Fire Accident

up to 100 cfm to the vent decay tanks. Backup is provided by rupture discs between primary cells which relieve the pressure to adjoining cells at approximately 4.5 psig. Cell shield plugs do not lift until approximately 6 psig is reached. Thus, the above spill can be adequately contained in the primary enclosures after the mechanical gas lock seals the sodium service cell.

(2) Nonradioactive Sodium Fire

All other sodium galleries may not be opened until 10 days after shutdown. At this time the sodium-24 activity has been reduced by a factor in excess of 10,000 by radiological decay and is negligible, leaving only the small amount of fission products to be considered. In the event that a fire occurs in these cells, the self-extinguishing characteristics caused by the formation of sodium-oxide limits the fire to 20 hr of burning and a burned depth of less than 1 ft. The initial burning rate is about 2.5 lb/hr-ft² with an average of 1.3 lb/hr-ft² over the 20-hr period.

In the nonradioactive sodium fire, it is assumed that the shield plug is removed from the primary IHX cell, and that a sodium spill results in burning of 44,000 lb of sodium in 20 hr. The sodium is assumed to have decayed for 10 days in accordance with the standard operating procedure pertaining to cell entry. As in the radioactive sodium fire, 10⁻⁴ of the core fission products are assumed trapped in the sodium at the time of the fire. Other assumptions are given in Table 16. The results of this accident are shown in Figure 16. Obviously, this accident places no additional requirements on the siting distances since doses are still below those of the fuel handling accident.

C. CONCLUSIONS

The accident analyses considered the most significant radiological accidents which could occur with an SGR at the AEC reference site. All other accidents considered resulted in lower radiation doses to the public. In each accident studied, a number of improbable failures were assumed to occur simultaneously or sequentially in order to create a potentially hazardous condition. The effects of each accident on the required siting distances are shown in Table 17.

The maximum credible accident (MCA) for this study is defined as "that accident whose off-site radiological consequences are not exceeded by any credible accident." The definition of credibility is subject to interpretation and

TABLE 16

ASSUMPTIONS USED TO CALCULATE ENVIRONMENTAL
DOSES FROM NONRADIOACTIVE SODIUM FIRES

Reactor operating time (days)	300
Reactor power (Mwt)	480
Cooling time (days)	10
Fraction of fission products in the coolant	10^{-4}
Coolant burning time (hr)	20
Coolant burning rate (avg burning rate)	1.3 lb/hr-ft ²
Surface area (ft ²)	1700
Total coolant burned (lb)	44,000
Stack height (meters)	60
(ft)	197
Stack flow rate (cfm)	200,000
No filters	
Release	

Significant Fission Products Plus Sodium	Release Fraction to Building	Release Fraction to Stack
halogens	1.0	0.5
noble gases	1.0	1.0
sodium	0.5	0.5

TABLE 17

REQUIRED SITING DISTANCES
(miles)

Accident	Exclusion Area	Low-Population Zone	Population Center
Core meltdown	0.06*	0.07	0.09
Sodium fire	0.06*	0.07	0.09
Fuel handling	0.06*	0.21	0.28

*The codes employed in the siting analysis do not apply
at distances less than 100 meters.

Figure 16. Nonradioactive Sodium Fire Accident

degree of probability for a given accident. The only accidents of those analyzed which are considered credible are the two sodium fires. The accident designated as the MCA is the fuel handling accident, although a credible fuel drop is considered to be only a small fraction of a fuel element.

This MCA requires a site boundary distance of less than 0.1 miles, which is within the limits of the reference site boundary. The 2-hr dose at the 0.4-mile site boundary is less than 10% of the permissible dose. Therefore, the exclusion area requirements are easily met by the reference site.

The low-population zone boundary distance required by the MCA is 0.21 mile. Since this distance falls within the site boundary, the low-population zone requirements are exceptionally well satisfied by the reference site. The population center distance required by the MCA is 0.28 mile, and also falls within the site boundary.

In summary, the results of a radiological hazards analysis for the prototype SGR located at the AEC reference site, indicate that it easily meets all present AEC safeguards requirements. In addition, it is apparent from the required siting distances that an SGR plant may be located at sites much closer to population centers than the hypothetical site.

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APPENDIX

A. PLANT DESIGN DATA

1. Reactor System

Thermal power	480 Mw
Active core diameter	13.5 ft
Active core height	14.0 ft
Axial reflector thickness	2.0 ft
Radial reflector thickness	2.0 ft
Cavity liner diameter	27.5 ft
Thermal shield thickness	4 in.
Reactor vessel liner diameter	19.6 ft
Reactor vessel diameter	19.5 ft
Calandria vessel diameter	19.5 ft

Fuel

Fuel material	UC
UC slug diameter	0.500 ± 0.005 in.
UC slug length	1 to 8 in.
UC slug stack height	14 ft ± 1/2 in.
Thermal bond	0.025 in. Na
Cladding material	304 SS
Cladding tube	0.570-in. OD 0.010-in. wall
Gas space length	40 in.
Number of rods	18
Central support tube	0.570-in. OD 0.030-in. wall
Process tube material	304 SS
Process tube	4.000-in. OD 0.025-in. wall
Square lattice pitch (at 68°F)	10.25 in.
Number of fuel elements	168
Number of control rods	24
Number of active core positions	196

Fuel per rod	6.88 kgU
Fuel per element	124 kgU
Core loading	20,830 kgU
Design maximum fuel slug diameter increase	0.040 in.
Design average fuel slug diameter increase	0.0265 in.
Design fission gas release fraction	3%
Design gas plenum pressure: end-of-life operating temperature	250 psig
Design maximum fuel clad strain	0.4%
Initial enrichment	3.80 wt % U ²³⁵
Final enrichment (average)	1.60 wt % U ²³⁵
Pu content of spent fuel (average)	6.86 gm of Pu/kg U
Average exposure	25,000 Mwd/MTU
Average specific power	23.0 kw/kg U
Average fuel power density	4.81 kw/in. ³
Average core power density	245 kw/ft ³
Average incore residence time (80% PF)	3.8 yr
Core heat transfer and coolant flow	
Coolant temperature rise hot-channel factor	1.04
Clad temperature hot-channel factor	1.11
Fuel temperature hot-channel factor	1.14
Element disadvantage factor	0.90
Axial peak-to-average power ratio	1.36
Radial peak-to-average power ratio	1.49
Overall peak-to-average power ratio	2.25
Average sodium velocity	12.6 ft/sec
Maximum sodium velocity	13.1 ft/sec
Coolant flow area	7.406 in. ²
Maximum element Na flow rate	33.8 lb/sec
Total sodium flow rate	13.55 x 10 ⁶ lb/hr
Core pressure drop	21 psi
Reactor thermal power output	480 Mwt
Heat generated outside fuel	6.0%
UC fuel thermal conductivity	13.3 $\frac{\text{Btu}}{\text{hr-ft-}^\circ\text{F}}$

Sodium bond thermal conductivity	$38.4 \frac{\text{Btu}}{\text{hr-ft-}^\circ\text{F}}$
Stainless steel-clad thermal conductivity	$12.7 \frac{\text{Btu}}{\text{hr-ft-}^\circ\text{F}}$
Average heat transfer film coefficient	$7090 \frac{\text{Btu}}{\text{hr-ft}^2\text{-}^\circ\text{F}}$
Nominal sodium inlet temperature	750° F
Nominal sodium outlet temperature	1150° F
Maximum hot-channel fuel temperature	1745° F
Maximum hot-channel inner clad temperature	1253° F
Maximum hot-channel outer clad temperature	1236° F
Maximum hot-channel Na outlet temperature	1212° F
Maximum nominal fuel temperature	1648° F
Maximum nominal clad temperature	1214° F
Mixed-mean sodium outlet temperature	1166° F
Maximum nominal linear rod power	24.0 kw/ft
Maximum nominal heat flux	548,800 Btu/hr-ft ²
Average nominal heat flux	243,750 Btu/hr-ft ²
Maximum nominal element channel power	4,257 kw

2. Reactor Control System

Total number of control rods	24
Rods with actuators and drives	24
Rods with scram latches	24
Reactivity control requirements; hot clean core	(% $\Delta k/k$)
Equilibrium xenon	2.3
Equilibrium samarium group of fission products	0.9
Xenon transient override	0.6
Burnup (7.5 mo, 80% PF)	5.6
Loss of sodium	3.5
Reactivity worth of control rods (% $\Delta k/k$)	15.0
Rod length and travel	
Poison column length	156 in.
Rod travel	156 in.

Uppermost rod position places bottom of poison column at core reflector interface	X
At full down position bottom of poison column is 1 ft above core bottom	X
Rod speed (nominal)	15 in./min
(minimum)	7.5 in./min
(maximum)	30 in./min
Reactivity rates	
Maximum allowable	0.033%/sec
Minimum required	0.0012%/sec
Nominal	0.005%/sec
Drive mechanism shall be torque limited	X
Any of the rods may be selected for automatic or manual control	X
Interlocks	
Interlock shall prevent the rods from being driven out of the core until all the scram latches have been engaged	X
No interlocks (except the lower limit switches) shall prevent the rods from being driven into the core	X
Rod response time	0.25 sec
Rod positioning and indication requirements	
Absolute position indication with respect to a frame of reference in the core	+1.0 in.
Accuracy of differential rod position requirements	+0.1 in.
Minimum rod motion possible by "inching" or jogging drive motors	0.05 in.
When motor is deenergized rod shall not move	X
Rod position indication does not have to follow scram but must be reset before restart	X
Scram-latch engagement shall be indicated	X
Full down position to be indicated coincidentally with reaching that position	X
If torque limit is provided by a slip clutch, slippage indication shall be provided	X

Limits and stops

An upper limit switch shall stop the drive motor and provide indication when the rod is at the upper most position	X
A physical stop shall prevent motion of more than 1/2 in. beyond the upper-most position	X
A lower limit switch shall stop the drive motor and provide indication when the rod is within 1/2 in. of the lowermost position	X
A physical stop shall prevent motion beyond the lower position	X
All switches used in instrumentation or control shall be adjustable in three axis	X
All switch installations shall be dual	X

Flexibility

All electrical maintenance shall be possible without reactor shutdown. The rod concerned shall be capable of being dropped when a scram occurs	X
Capability shall be provided to test each rod circuit and rod drive without the poison column, provided the reactor is shut down	X

Safety system requirements

Scram delay time	0.5 sec
------------------	---------

Rod drop characteristics

Average minimum rod fall acceleration	0.1 g
Rods shall be capable of being dropped from any position in the core	X

Safety duty cycle requirements

Average number of rod drops per year	50
Maximum number of rod drops per year	150
Maximum number of complete withdrawals	1000

Shim duty cycle requirements

Number of cycles per year	3300/A
Amplitude of cycles	$0.05 < A < 2.5$ in.

Source

Source material (capsule)	Sb_2O_4
Source material (cylinder)	Beryllium
Source strength	1×10^9 n/sec
Operating temperature	1300°F
Containment thimble	304 SS
Required	1

3. Sodium Heat Transfer System

Primary Loop	Main Circuit	Auxiliary Circuit
Number of loops	1	1
Capacity (nominal Mwt)	432	48
Sodium temperature (°F)		
Reactor inlet	750	750
outlet	1150	1150
Sodium flow rate (lb/hr)	12.2×10^6	1.35×10^6
Pump capacity (gpm)	30,700	3,400
Total dynamic head (ft)	190	190
NPSH available	28	28
Motor power (hp)	2000	2000
Pipe sizes (in.)		
Suction	30	12
Discharge	28	10
Cold-leg	26	10
Secondary Loop		
Number of loops	1	1
Capacity (nominal Mwt)	432	48
Sodium temperature (°F)		
Steam generator inlet	1075	1075
Steam generator outlet	675	675
Sodium flow rate (lb/hr)	12.2×10^6	1.35×10^6
Pump capacity (gpm)	30,700	3,400
Total dynamic head (ft)	190	190
Motor power (hp)	2,000	250

	<u>Main Circuit</u>	<u>Auxiliary Circuit</u>
Pipe sizes (in.)		
Hot-leg	28	10
Cold-leg	26	10

Intermediate Heat Exchanger

Type of construction	Modular shell and tube	Modular shell and tube
Number of units	2	1
Duty per unit (Btu/hr)	7.4×10^8	1.66×10^8
Δt drive ($^{\circ}\text{F}$)	75	75
Surface area per unit (ft^2)	10,900	2,425

	<u>Tube Side</u>	<u>Shell Side</u>	<u>Tube Side</u>	<u>Shell Side</u>
Fluid (sodium)	Primary	Secondary	Primary	Secondary
Flow (lb/hr) per unit	6.1×10^6	6.1×10^6	1.35×10^6	1.35×10^6
Temperature in ($^{\circ}\text{F}$)	1150	675	1150	675
out ($^{\circ}\text{F}$)	750	1075	750	1075
Pressure drop (psi)	7	4	7	4
Material (stainless steel) 304	304	304	304	304

4. Steam Generator

Evaporator (low temperature module)

Type construction	Shell and tube	Shell and tube
Total number of modules	26	3
Number of banks	2	1
Number of modules per bank	13	3
Duty per module (Btu/hr)	31.5×10^6	36.5×10^6
Surface area per module (ft^2)	476	433

	<u>Tube Side</u>	<u>Shell Side</u>	<u>Tube Side</u>	<u>Shell Side</u>
Fluid	Water/steam	Secondary Na	Water/steam	Secondary Na
Flow per module (lb/hr)	45,800	459,600	53,200	443,000
Temperature in ($^{\circ}\text{F}$)	473	900	473	945
out ($^{\circ}\text{F}$)	695	675	695	675
Inlet feedwater pressure (psia)	2,700	-	2,700	-
Material	5 Cr-1/2 Mo-Ti	5 Cr-1/2 Mo-Ti	5 Cr-1/2 Mo-Ti	5 Cr-1/2 Mo-Ti

Superheater (High Temperature Module)

	<u>Main Circuit</u>	<u>Auxiliary Circuit</u>
Type of construction	Shell and tube	Shell and tube
Total number of modules	16	2
Number of banks	2	1
Number of modules per bank	8	2
Duty per module (Btu/hr)	23.7×10^6	25.5×10^6
Surface area per module (ft ²)	550	484

	<u>Tube Side</u>	<u>Shell Side</u>	<u>Tube Side</u>	<u>Shell Side</u>
Fluid	Steam	Secondary Na	Steam	Secondary Na
Flow per module	74,500	453,100	79,700	664,000
Temperature in (° F)	695	1075	695	1075
out (° F)	1005	900	1005	945
Outlet steam pressure (psia)	2,500	-	2,500	-
Material (stainless steel)	321	321	321	321

Reheater Module

Type construction	Shell and tube	None required
Total number of modules	6	
Number of banks	2	
Number of modules per bank	3	
Duty per module (Btu/hr)	41.1×10^6	
Surface area per module (ft ²)	2,030	

	<u>Tube Side</u>	<u>Shell Side</u>
Flow per module (lb/hr)	197,300	783,300
Temperature in (° F)	630	1075
out (° F)	1005	900
Outlet steam pressure (psia)	500	-
Material (stainless steel)	321	321

5. Turbine Generator

Rated generator gross electrical output, rated (kw)	210,049
Ultimate generator gross electrical output, maximum (kw)	221,000

Net turbine cycle heat rate at rated capability	7,830 Btu/kwh
Net turbine cycle heat rate of ultimate capability	7,844 Btu/kwh
Throttle steam flow at rated capability	1,351,741 lb/hr
Reheat steam flow at rated capability	1,183,827 lb/hr
Condenser steam flow at rated capability	923,560 lb/hr
Main steam pressure	2400 psig
Reheat steam pressure	484.8 psig
Initial throttle and reheat steam temperature	1000° F
Turbine exhaust pressure (in. Hg abs)	1-1/2
Number of stages of feedwater heater	7
Final feedwater temperature	472.5° F
Generator kva rating at 45 psig H ₂	256,000
Generator preferred terminal voltage	22.0 kv
Power factor	0.85
Short-circuit ratio	0.64
Preferred field voltage (v)	375

6. Condensing System

Main Condenser

Operating pressure	1-1/2 in. Hg abs
Steam condensed	923,560 lb/hr
Condenser duty	876,000,000 Btu/hr
Entering circulating water temperature	57° F
Leaving circulating water temperature	77° F
Circulating water flow	87,600 gpm
Surface required	68,500 ft ²
Tube OD, gauge, length	7/8 in. tube 18 BWG 38-ft length
Single-pass condenser, divided water box, horizontal unit	
Circulating water tube velocity	7.5 ft/sec
Cleanliness factor	0.85
Tube material	Admiralty

Circulating Water Pumps (two required)

Data per Circulating Pump

Water flow	43,800 gpm
Pump total dynamic head	25 ft
Pump efficiency	85%
Motor rating	300 hp, 235 rpm
Inlet water temperature	57° F
Condensate pumps; two required, horizontal centrifugal	

Data per Condensate Pump

Condensate flow	1,550 gpm
Pump total dynamic head	820 ft
Pump efficiency	80%
Motor rating	400 hp, 1,800 rpm
Inlet water temperature	92° F
Circulating water piping - two pipes, from each circulating water pump at 48-in. OD each	

7. Feedwater System

Main boiler feedwater pumps: (two required:
2 - 50%)

Feedwater flow, each pump	1550 gpm
Pump total dynamic head	7,500 ft
Pump efficiency	81%
Pump rpm	7,000
Inlet feedwater temperature	315° F
Barrel type - 6-stage electric motor hp	4,000

High Pressure Feedwater Heater No. 1

Feedwater flow (tube side)	1,351,741 lb/hr
Steam flow (shell side)	119,053 lb/hr
Shell steam operating pressure	527.3 psia
Shell steam operating temperature	472.5° F
Feedwater temperature in	396.6° F
Feedwater temperature out	472.5° F

Feedwater pressure in	2,700 psig
Tube material - carbon steel	Total surface, 6,700 ft ²
Design pressure	Shell - 650 psig Tube - 3,000 psig
Design temperature	Shell - 700°F Tube - 500°F

High Pressure Feedwater Heater No. 2

Feedwater flow	1,351,741 lb/hr
Steam flow	52,432 lb/hr
Shell steam pressure	224.5 psia
Shell steam temperature	391.6°F
Feedwater temperature in	351.7°F
out	396.6°F
Tube material - carbon steel	Total surface, 5,600 ft ²
Design pressure	Shell - 300 psig Tube - 3,000 psig
Design temperature	Shell - 550°F Tube - 475°F

High Pressure Feedwater Heater No. 3

Feedwater flow	1,351,741 lb/hr
Steam flow	37,169 lb/hr
Shell steam pressure	137.6 psia
Shell steam temperature	351.7°F
Feedwater temperature in	318.5°F
out	351.7°F
Feedwater pressure in	2,750 psig
Tube material - carbon steel	Total surface, 4,750 ft ²
Design pressure	Shell - 225 psig Tube - 3,000 psig
Design temperature	Shell - 500°F Tube - 450°F

Deaerating Heater No. 4

Oxygen limit	0.005 cc/liter
Entering condensate flow	1,072,997 lb/hr
Entering condensate temperature	244.3°F

Entering drains flow	208,654 lb/hr
Entering drains temperature	328.5° F
Entering steam pressure	81.5 psia
Entering steam temperature	313.2° F
Entering steam flow	70.090 lb/hr
Leaving feedwater flow	1,351,741 lb/hr
Leaving feedwater pressure	81.5 psia
Leaving feedwater temperature	313.2° F
Shell design pressure	150 psig
Shell design temperature	400° F

Low Pressure Heater No. 5

Feedwater flow	1,072,997 lb/hr
Steam flow	31,386 lb/hr
Shell steam pressure	29.5 psia
Shell steam temperature	249.5° F
Feedwater temperature in	214.2° F
out	244.3° F
Feedwater pressure in	110 psia
Tube material - Admiralty	Total surface, 4,675 ft ²
Design pressure	Shell - 150 psig Tube - 200 psig
Design temperature	Shell - 400° F Tube - 400° F

Low Pressure Heater No. 6

Feedwater flow	1,072,997 lb/hr
Steam flow	31,779 lb/hr
Shell steam pressure	16.9 psia
Shell steam temperature	219.2° F
Feedwater temperature in	183.0° F
Feedwater temperature out	214.2° F
Feedwater pressure in	110 psig
Tube material - Admiralty	Total surface, 5,450 ft ²
Design pressure	Shell - 100 psig Tube - 200 psig

Design temperature	Shell - 300° F Tube - 350° F
Low Pressure Feedwater Heater No. 7	
Feedwater flow	926,910 lb/hr
Steam flow	82,923 lb/hr
Shell steam pressure	8.82 psia
Shell steam temperature	187.3° F
Feedwater temperature in	92.5° F
out	182.3° F
Feedwater pressure in	110 psig
Tube material - Admiralty	Total surface, 3,375 ft ²
Design pressure	Shell - 50 psig Tube - 200 psig
Design temperature	Shell - 250° F Tube - 300° F

EVERY EVEN PAGE NUMBER IS INTENDED TO BE BLANK
FOR THE REMAINDER OF THIS DOCUMENT. STARTING
WITH THIS PAGE.

Drawing 1. Plot Plan
(L159-FA001)

NAA-SR-9213

Drawing 2. Plant Arrangement,
Reactor Building, Plan View
at Elevation 112 ft-0 in.
(L159-GA001)
NAA-SR-9213
175

Drawing 3. Plant Arrangement,
Reactor Building, Plan View
at Elevation 95 ft-0 in.
(L159-GA002)

Drawing 4. Plant Arrangement,
Reactor Building, Plan View
at Elevation 75 ft-0 in.
(L159-GA003)

Drawing 5. Plant Arrangement,
Reactor Building Section A-A
(L159-GA004)

NAA-SR-9213
181

Drawing 6. Plant Arrangement,
Reactor Building Section B-B
(L159-GA005)

SECTION C-C

Drawing 7. Plant Arrangement,
Reactor Building Section C-C
(L159-GA006)

Drawing 8. Plant Arrangement,
Turbine Building Plans
(L159-GA007)
NAA-SR-9213
187

Drawing 9. Plant Arrangement,
Turbine Building Sections
(L159-GA008)

NOTES UNLESS OTHERWISE SPECIFIED
1. FOR DETAILS SEE A1 DWG L159 RA002

Drawing 10. Reactor General
Arrangement
(L159-RA001)
NAA-SR-9213
191
DTI 10-0000 1948

DETAIL A
REF A1 DWS L159-R5001

DETAIL B
REF A1 DRAWING L159-R5001

WELD TESTING SYMBOLS	
SYMBOL	TYPE OF TEST
PT	PERMEANT INSPECTION
HT	HE LEAK TEST
RT	RADIOGRAPHIC INSPECTION
UT	ULTRASONIC INSPECTION
(EF)	INSPECT ROOT & FINAL PASS
(IF)	INSPECT FINAL PASS ONLY

Drawing 11. Reactor General
Arrangement Details
(L159-RA002)

NAA-SR-9213
193

Drawing 12. Loading Face Shield
(L159-RL001)
NAA-SR-9213
195

NOTES UNLESS OTHERWISE SPECIFIED
FOR GENERAL NOTES SEE A1
DRAWING L159 RL001

Drawing 13. Loading Face
Shield Details
(L159-RL002)
NAA-SR-9213
197

VIEW I⁺ FULL SIZE
REF A1 DWG L159-RL001

VIEW H FULL SIZE
REF A1 DWG L159-RL001

VIEW J FULL SIZE
REF A1 DRAWING L159-RL001

VIEW G HALF SIZE
REF A1 DWG L159-RL001

SECTION K-K FULL SIZE
REF A1 DWG L159-RL001

NOTES UNLESS OTHERWISE SPECIFIED
1 FOR GENERAL NOTES SEE A 1,
DRAWING L159-RL001

Drawing 14. Loading Face
Shield Details
(L159-RL003)

NAA-SR-9213
199

WELD TESTING SYMBOLS	
SYMBOL	TYPE OF TEST
PT	PENETRANT INSPECTION
HT	HE LEAK TEST
RT	RADIOGRAPHIC INSPECTION
UT	ULTRASONIC INSPECTION
A(R (F)	INSPECT ROOT & FINAL PASS INSPECT FINAL PASS ONLY

Drawing 15. Cavity Liner and
Thermal Shield
(L159-RS005)
NAA-SR-9213
201
OSTI ID: 000019148

Drawing 16. Reactor Vessel
(L159-RS001)
NAA-SR-9213
203

NOTES: UNLESS OTHERWISE SPECIFIED
 ALL MATERIAL TO COMPLY WITH THE APPLICABLE ASTM SPECIFICATION FOR TYPE 304 STAINLESS STEEL, 0.06 MIN CARBON CONTENT
 (2) ULTRASONIC TEST PER A1 SPEC NL159 RS001
 (3) FABRICATE PER A1 SPEC NL159 RS001
 (4) EXACT NO & LOCATION OF JOINTS TO BE DETERMINED BY SUPPLIER
 (5) ALL UNMACHINED INSIDE SURFACES OF THE LINER TO HAVE A SURFACE ROUGHNESS OF 250 RMS MAX
 (6) DURING LINER FABRICATION THE REACTOR VESSEL IS TO BE USED AS A SUPPORTING FORM

WELD TESTING SYMBOLS	
SYMBOL	TYPE OF TEST
* PT	PENETRANT INSPECTION
HT	He LEAK TEST
RT	RADIOGRAPHIC INSPECTION
UT	ULTRASONIC TEST
* (RF)	INSPECT ROOT & FINAL PASS
(F)	INSPECT FINAL PASS ONLY

Drawing 17. Reactor Vessel Liner
(L159-RS003)

WELD TESTING SYMBOLS	
SYMBOL	TYPE OF TEST
*PT	PENETRANT INSPECTION
HT	H ₂ LEAK TEST
RT	RADIOGRAPHIC INSPECTION
UT	ULTRASONIC INSPECTION
*(RF)	INSPECT ROOT & FINAL PASS
(F)	INSPECT FINAL PASS ONLY

NAA-SR-9213
207

Drawing 19. Outlet Manifold
(L159-RS004)

NAA-SR-9213
209

DETAIL "D"
REF. A.I. DRAWING - L159-R5001

DETAIL "E"
REF A.I. DRAWING L139-R5001

NAA-SR-9213
211

- NOTES UNLESS OTHERWISE SPECIFIED
 1. ALL MAT'L TO COMPLY WITH THE APPLICABLE ASTM SPEC FOR TYPE 304 STAINLESS STEEL 0.06 MIN CARBON CONTENT
 2. ULTRASONIC TEST PER A1 SPEC NL159 R5001
 3. FABRICATE PER A1 SPEC NL159 R5001
 4. ALL UNMACHINED SURFACES TO HAVE A SURFACE ROUGHNESS OF 250 RMS MAX
 5. FOR DETAILS SEE A1 DRAWINGS L159 RC 002 & L159-RC 003

CALANDRIA ASSEMBLY SEQUENCE

1. ASSEMBLE 'Y' RING FROM PRE-FAB SHOP SECTIONS SO THAT OD MATCHES ELLIPSE HEAD OD
2. ASSEMBLE BOTTOM PLATE FROM PRE-FAB SHOP SECTIONS TRIM OD TO MATCH ID OF 'Y' RING AND WELD THIS JOINT
3. JOIN ELLIPSE HEAD TO 'Y' RING AND BOTTOM PLATE ASSY
4. INSTALL SUPPORT STAYS BETWEEN BOTTOM PLATE AND ELLIPSE HEAD TO PROVIDE STABILITY TO BOTTOM PLATE
5. INSTALL BOTTOM JOINT TO BOTTOM PLATE BY WELDING. ALIGN NOZZLES
6. LOCATE AND WELD SUPPORT PEDESTALS FOR GRAPHITE FILLER LOGS USE LOCATING TEMPLATE IF NECESSARY
7. STARTING AT 'Y' RING, BUILD UP THE VESSEL WALL TO THE JOINT AT REF ELEV 73 FT 8 3/8 IN (DETAIL H)
8. ASSEMBLE THE KNUCKLE RADIUS SECTION FROM PRE-FAB SHOP SECTIONS AND JOIN TO VESSEL WALL
9. ASSEMBLE TOP PLATE FROM PRE-FAB SHOP SECTIONS. TRIM OD TO MATCH ID OF KNUCKLE RADIUS SECTION AND ACHIEVE HOLE ALIGNMENT WITH THE BOTTOM PLATE. ATTACH ALL HANGER ROD CONNECTIONS TO THE TOP SIDE AND ALIGNMENT PEDESTALS FOR THE GRAPHITE FILLER LOGS ON THE BOTTOM SIDE. JOIN TOP PLATE TO KNUCKLE RADIUS SECTION
10. WELD CALANDRIA VESSEL SUPPORT LUGS AT TOP PERIPHERY OF VESSEL WALL
11. LOCATE AND WELD LINK BAR KEYWAYS TO VESSEL WALL
12. ALIGN AND WELD TOP JOINT. AFTER CALANDRIA IS INSTALLED IN REACTOR VESSEL AND TOP PLATE HANGER RODS HAVE BEEN CONNECTED AND ADJUSTED
13. AFTER WELDING, TOP & BOTTOM JOINTS (STEPS 5 & 12) TO BE WITHIN ± 3/64\"

WELD TESTING SYMBOLS	
SYMBOL	TYPE OF TEST
* PT	PENETRANT INSPECTION
HT	He LEAK TEST
RT	RADIOGRAPHIC INSPECTION
UT	ULTRASONIC INSPECTION
*(RF)	INSPECT ROOT & FINAL PASS
(F)	INSPECT FINAL PASS ONLY

TOP PLATE HANGER ROD CONNECTION
DETAIL **D**
REF AI DWG L159-RC001

VESSEL SUPPORT LUG
DETAIL **J**
REF AI DWG L159-RC001

LINK BAR KEYWAY
DETAIL **K**
REF AI DWG L159-RC001

SUPPORT STAY ATTACHMENT
DETAIL **L**
REF AI DWG L159-RC001

NOZZLE REINFORCEMENT
DETAIL **F** 1/4 SIZE
REF AI DWG L159-RC001

RADIUS SECTION
DETAIL **H**
REF AI DWG L159-RC001

T RING
DETAIL **G**
REF AI DWG L159-RC001

NOTES: UNLESS OTHERWISE SPECIFIED
1- FOR GENERAL NOTES SEE
AI DRAWING L159-RC001.

WELD	TESTING SYMBOLS
SYMBOL	TYPE OF TEST
PT	PENETRANT INSPECTION
HT	HE LEAK TEST
RT	RADIOGRAPHIC INSPECTION
UT	ULTRASONIC TEST
(RF)	INSPECT ROOT & FINAL PASS
(F)	INSPECT FINAL PASS ONLY

Drawing 22. Calandria Details
(L159-RC002)

NAA-SR-9213

NOTES UNLESS OTHERWISE SPECIFIED
1 FOR GENERAL NOTES SEE A I.
DRAWING L159-RC001

WELD TESTING SYMBOLS	
SYMBOL	TYPE OF TEST
* PT	PENETRANT INSPECTION
HT	HE LEAK TEST
RT	RADIOGRAPHIC INSPECTION
UT	ULTRASONIC INSPECTION
*(RT)	INSPECT ROOT & FINAL PASS
(CF)	INSPECT FINAL PASS ONLY

Drawing 23. Calandria Details
(L159-RC003)

NAA-SR-9213
217

- NOTES UNLESS OTHERWISE SPECIFIED
- 1 TAG (ONLY) FOR IDENTITY WITH PART NO. AND LATEST REVISION LETTER
 - 2 HANDLING AND MACHINING REQUIREMENTS ARE LISTED BELOW:
 - (A) STORE GRAPHITE ON CLEAN PAPER OR OTHER CLEAN SURFACE. DO NOT PLACE DIRECTLY ON THE FLOOR
 - (B) ANY LUBRICATING OIL, CUTTING OILS, OR FOREIGN MATERIALS MUST BE PREVENTED FROM COMING IN CONTACT WITH THE GRAPHITE
 - 3 FABRICATE PER AT SPEC NL159-R5001
 - ④ 4 MACHINING OF SLOTS TO BE PERFORMED AFTER ALL OTHER MACHINING IS COMPLETE

Drawing 24. Upper Graphite Log,
Process Tube
(L159-RC008)

Drawing 25. Lower Graphite Log,
Process Tube
(L159-RC009)

- NOTES: UNLESS OTHERWISE SPECIFIED
- 1 TAG (ONLY) FOR IDENTITY WITH PART NO AND LATEST REVISION LETTER
 - 2 HANDLING AND MACHINING REQUIREMENTS ARE LISTED BELOW
 - (A) STORE GRAPHITE ON CLEAN PAPER OR OTHER CLEAN SURFACE DO NOT PLACE DIRECTLY ON THE FLOOR
 - (B) ANY LUBRICATING OIL, CUTTING OILS, OR FOREIGN MATERIALS MUST BE PREVENTED FROM COMING IN CONTACT WITH THE GRAPHITE
 - 3 FABRICATE FOR AT SPEC NLIS-ES001
 - ① 4 MACHINING OF SLOTS TO BE PERFORMED AFTER ALL OTHER MACHINING IS COMPLETE

- NOTES UNLESS OTHERWISE SPECIFIED
- 1 TAG (ONLY) FOR IDENTITY WITH PART NO AND LATEST REVISION LETTER
 - 2 HANDLING AND MACHINING REQUIREMENTS ARE LISTED BELOW
 - (A) STORE GRAPHITE ON CLEAN PAPER OR OTHER CLEAN SURFACE DO NOT PLACE DIRECTLY ON FLOOR
 - (B) ANY LUBRICATING OIL, CUTTING OILS, OR FOREIGN MATERIALS MUST BE PREVENTED FROM COMING IN CONTACT WITH THE GRAPHITE
 - 3 FABRICATE PER AI SPEC NL159 25001
 - ① 4- LENGTH TO BE DEPENDENT UPON NO OF PIECES NECESSARY TO MAKE COMPLETED LOG 17' 3" LG
 - ② 5- MAX DEVIATION OF HOLE ϕ FROM ϕ OF LOG TO BE $\pm 1/8$

Drawing 27. Graphite Log,
Reflector
(L159-RC016)
NAA-SR-9213
225

NAA-SR-9213
227

SECTION A - A

NOTES: UNLESS OTHERWISE SPECIFIED

- 1 TAG (ONLY) FOR IDENTITY WITH PART NO AND LATEST REVISION LETTER
- 2 HANDLING AND MACHINING REQUIREMENTS ARE LISTED BELOW
 - (A) STORE GRAPHITE ON CLEAN PAPER OR OTHER CLEAN SURFACE DO NOT PLACE DIRECTLY ON THE FLOOR
 - (B) THE MACHINE ON WHICH THE GRAPHITE IS MACHINED MUST BE CLEAN, WITH TABLE, CUTTING HEAD, AND CUTTER, FREE OF ANY DIRT, OILS, COOLANTS OR OTHER FOREIGN MATERIALS
 - (C) ANY LUBRICATING OIL DRIP OR SPLASH MUST BE PREVENTED FROM COMING IN CONTACT WITH THE GRAPHITE
 - (D) NO CUTTING FLUIDS OR COOLANTS ARE TO BE USED IN MACHINING THE GRAPHITE
 - (E) SMALL GOUGE, SCRIBED NUMBER, SCRATCHES, OR OTHER MINOR SURFACE BLEMISHES WILL BE ACCEPTABLE PROVIDING THEY DO NOT EXCEED .005 INCH DEEP BY 1/2 INCH WIDE BY ONE INCH LONG

① 5 TO BE PARALLEL TO EACH OTHER WITHIN .003 T.I.R.

NOTES: UNLESS OTHERWISE SPECIFIED
1. FABRICATE PER A1 SPEC NL 159-R3001

WELD TEST SYMBOLS	
SYMBOL	TYPE OF TEST
* PT	PENETRANT TEST
RT	RADIOGRAPHIC INSPECT
HT	HE LEAK TEST
UT	ULTRASONIC TEST
* (RF) (F)	INSPECT ROOT & FINAL PASS INSPECT FINAL PASS ONLY

Drawing 30. Calandria Assembly
(L159-RC017)

NAA-SR-9213
231

- NOTES UNLESS OTHERWISE SPECIFIED
1. FABRICATE PER AI SPEC NL159-R5001.
 2. "RUBBER STAMP" DWS NO. 8 LATEST REVISION
LETTER FOR IDENTIFICATION
 3. WELD PER ASME BOILER & PRESSURE
VESSEL CODE SECTION VIII NO CODE STAMP
REQD
 - ① 4. FINAL WELD
 - ② 5. HELIUM LEAK CHECK PER AI SPEC
NL159-R5001 BY PRESSURE METHOD

1	L159-RC014	Bottom Joint		
1	L159-RC011	Top Joint		
1	L159-RC008	Lower Graphite Log		
1	L159-RC010	Guide Assy		
1	L159-RC007	Belows Assy		
1	L159-RC009	Upper Graphite Log		
5	L159-RC007	Graphite Log		
1	L159-RC013	Tube	ASME SECTION VIII NO. 1	
REV	DATE	DESCRIPTION	MATERIAL	SPEC

Drawing 31. Process Tube Assembly
(L159-RC006)

NAA-SR-9213

233

WELD TESTING SYMBOLS	
SYMBOL	TYPE OF TEST
* PT	PENETRANT INSPECTION
HT	HE LEAK TEST
RT	RADIOGRAPHIC INSPECTION
UT	ULTRASONIC INSPECTION
* (RF) (F)	INSPECT ROOT & FINAL PASS INSPECT FINAL PASS ONLY

NAA-SR-9213
235

WELD TEST SYMBOLS	
SYMBOL	TYPE OF TEST
R PT	PENETRANT INSPECTION
HT	He LEAK TEST
ET	RADIOGRAPHIC INSPECTION
UT	ULTRASONIC INSPECTION
* (RF) (F)	INSPECT ROOT & FINAL PASS INSPECT FINAL PASS ONLY

Drawing 33. Shield Plug, Fuel
(L159-RL005)
NAA-SR-9213
237
OST ID: OOOO 19148

- NOTES: UNLESS OTHERWISE SPECIFIED
- 1 FABRICATE PER A.I. SPEC NL159-RS001.
 - 2 BREAK SHARP CORNERS .010 - .020 R.
 - (1) 3 DIAS TO BE CONCENTRIC WITHIN .005 TIR
 - (2) 4 BREAK SHARP CORNERS .020 - .030 R
 - 5 ALL MAT'L TO COMPLY WITH APPLICABLE ASTM SPEC FOR TYPE 304 STAINLESS STEEL, .006 MIN. CARBON CONTENT.

WELD TEST SYMBOLS	
SYMBOL	TYPE OF TEST
*DT	PENETRANT TEST
RT	RADIOGRAPHIC TEST
HT	HELIUM LEAK TEST
UT	ULTRASONIC TEST
*(RF)	ROOT & FINAL PASS
(F)	FINAL PASS ONLY

Drawing 34. Top Joint Calandria
(L159-RC004)

NAA-SR-9213
239

NOTES: UNLESS OTHERWISE SPECIFIED

1. FABRICATE PER AI SPEC NL159-RS001.
- ① 2. BREAK SHARP EDGE .020-.030 R.
3. MATERIAL TO COMPLY WITH THE APPLICABLE ASTM SPEC FOR TYPE 304 SST 0.06 MIN. CARBON CONTENT.

Drawing 35. Bottom Joint Calandria
(L159-RC005)

NAA-SR-9213
241

- NOTES: UNLESS OTHERWISE SPECIFIED
1. FABRICATE PER AS SPEC NL159-RS001
 - ① 2. DIA TO BE CONCENTRIC TO EACH OTHER WITHIN .005 T.I.R.
 3. MATERIAL TO BE TYPE 304 STAINLESS STEEL PER APPLICABLE ASTM SPEC. 0.06 MIN CARBON CONTENT

WELD TEST SYMBOLS	
SYMBOL	TEST
* PT	PENETRANT TEST
HT	He LEAK TEST
RT	RADIOGRAPHIC INSPECT
UT	ULTRASONIC TEST
√(R,F)	ROOT & FINAL PASS
·(F)	FINAL PASS ONLY.

- NOTES: UNLESS OTHERWISE SPECIFIED:
 1. FABRICATE PER AI SPEC NL159-R3001.
 ① 2. DIA'S TO BE CONCENTRIC WITHIN .005 T.I.R.

-3 JOINT

-4 RING

- NOTES: UNLESS OTHERWISE SPECIFIED
1. FABRICATE PER AI SPEC NL159-RS001.
 2. HELIUM LEAK CHECK PER AI SPEC NL159-RS001 BY PRESSURE METHOD
 3. SUPPLIER TO SUPPLY TO AI DEPT 717-721 FABR DNGS & AI DESIGN SHEETS (6 COPIES EACH) FOR -5 BELLWS FOR REVIEW PRIOR TO FABRICATION
 4. DESIGN CONDITIONS FOR BELLWS:
 - (A) AXIAL DEFLECTION 0.6 LB
 - (B) DESIGN PRESSURE 6 PS I.
 - (C) DESIGN TEMP 1200°F
 - (D) SPRING RATE (MAX) 280 LB PER INCH
 - (E) MIN WALL THICKNESS 0.020

QTY	PART NO.	DESCRIPTION	QUANTITY	SPEC	UNIT
1	-5	BELLWS		557L TP316	
1	-4	RING	3/8 (STK)	557L TP304	
1	-3	RING	1/2 (STK)	557L TP304	
LIST OF MATERIAL					

Drawing 38. Bellows Assembly
(L159-RC011)
NAA-SR-9213
247

- NOTES: UNLESS OTHERWISE SPECIFIED
1. FABRICATE PER AI SPEC NL159-RS001.
 2. ALL MATERIAL TO COMPLY TO APPLICABLE ASTM SPEC FOR TYPE 304 STAINLESS STEEL. 0.06 MIN CARBON CONTENT.
 - ① 3. MAX ALLOWABLE OUT OF ROUNDNESS TO BE WITHIN .005 TIR.

WELD TEST SYMBOLS	
SYMBOL	TYPE OF TEST
*PT	PENETRANT TEST
HT	He LEAK TEST
RT	RADIOGRAPHIC INSPECT
UT	ULTRASONIC TEST
* (R F)	INSPECT ROOT & FINAL PASS
(F)	INSPECT FINAL PASS ONLY

Drawing 39. Guide Assembly
(L159-RC010)

NAA-SR-9213

1 ALL STEEL PARTS SHALL BE THOROUGHLY
CLEANED PRIOR TO ASSEMBLY BY
WIPING WITH A CLEAN ACETONE-SATURATED
LINT FREE CLOTH THE INTERNAL &
EXTERNAL SURFACES SHALL BE FREE OF
GREASE, OIL DIRT OR OTHER VISIBLE CONTAMINANTS

NOTES: UNLESS OTHERWISE SPECIFIED

Drawing 41. Fuel Element Assembly
(L159-RF001)

NAA-SR-9213

253

OSTI.D:0000 19/48

Drawing 42. Control Rod Layout
(L159-RM001)

NAA-SR-9213

255

OSTID:000019148

BRIEF FUEL HANDLING SCHEDULE

RECEIVE NEW FUEL INTO REACTOR ROOM
 BUILDING HOIST MOVES NEW FUEL TO NEW FUEL STORAGE
 AND PREPARATION ROOM
 BUILDING HOIST MOVES NEW FUEL TO PICK-UP CELL
 AFTER INSPECTION.
 FUEL HANDLING MACHINE PICKS UP NEW FUEL AT PICK-UP
 CELL AND TAKES FUEL TO REACTOR
 NEW FUEL IS EXCHANGED FOR SPENT FUEL
 FUEL HANDLING MACHINE MOVES SPENT FUEL TO WET
 FUEL STORAGE
 AFTER DECAY FUEL HANDLING MACHINE TAKES SPENT FUEL
 FROM WET STORAGE TO THE SHIPPING CASK
 WHEN FILLED BUILDING HOIST MOVES SHIPPING CASK TO
 SHIPPING TRUCK

Drawing 43. Fuel Handling Schematic
 (L159-HA001)

Drawing 44. Fuel Handling Machin
 (L159-HA002)
 NAA-SR-9213
 259

GENERAL DATA	GANTRY	TROLLEY
GROSS WT (LOADED) (ESTIMATE)	525 KIPS	405 KIPS
SHIPPING WT. (ESTIMATE)	110 KIPS	55 KIPS
MAX. WHL. LOAD (ESTIMATE)	103 KIPS	110 KIPS
SPEED RANGE	1/4 - 90 FPM	1/4 - 30 FPM

EXCEPT FOR DIMENSIONS MARKED *
ALL DIMENSIONS ARE APPROX.

Drawing 45. Fuel Handling Machine
Gantry and Trolley
(L159-HA005)

NAA-SR-9213
261

REFERENCE LIST	
AT DWG	7518 44516 ASSY FUEL GRAPPLE
	7518 44525 ASSY DOUBLE VALVE
	7518 44523 SEAL ASSY
	7518 44525 TAPE FUEL HANDLING HOIST
	7518 44523 GUIDE TUBE DETAILED ASSY
	7518 44525 DETAILS CASK
	7518 44525 DETAILS GRAPPLE TURNBUCKLE
AT DWG	L159 HA004 FUEL HANDLING CASK DETAILS
	200 MME 5GR (REVISED A.D.O.F.E.E.)
TOR NO	NAA SR 8058 FUEL HANDLING HEAT TRANSFER ANALYSIS
	2 1/2 HAMMER, 2 EX 02
NAA SR 8069	CURRENT ASFR FUEL HANDLING
	HOW BE OVER 2 & 63
NAA SR 8081	200 MME FUEL HANDLING CONCEPT
	C.M. MEM
TOR NO	NAA SR 8766 FUEL ELEMENT GRAPPLE & HOLD DOWN LATCH DEVELOPMENT
	10 M. VERNIA

GRAPPLE - SHIELD PLUG & CONTROL ROD
SCALE: FULL
NOTE: THE BASIC DESIGN CRITERIA OF THIS GRAPPLE IS TYPICAL TO THE FUEL ELEMENT GRAPPLE. (REF. AS DWG. 7508-04516)

SECTION I-E
SCALE: 1/2

VIEW H
SCALE: 1/2

SECTION C-C
SCALE: 1/2
(SHOWS GAS LOCK IN FULL UP POSITION)

SECTION I-I
SCALE: 1/2

SECTION A-A
SCALE: 1/2

SECTION B-B
SCALE: 1/2

SECTION F-F
SCALE: FULL
(SHOWS GAS LOCK IN FULL UP POSITION)

SECTION B-B

TOP VIEW

SECTION A-A

Drawing 48. Fuel Storage Assembly
(L159-HP003)

NAA-SR-9213

267

OSTI ID: OOOO19148

Drawing 49. Fuel Storage
Cavity Liner
(L159-HP004)
NAA-SR-9213
269

NOTES UNLESS OTHERWISE SPECIFIED
 1 FABRICATE PER AS SPEC
 2 ALL WELDS TO COMPLY WITH SECTION
 VIII OF THE ASME BOILER & PRESSURE
 VESSEL CODE

WELD TESTING SYMBOL	
SYMBOL	TYPE OF TEST
* PT	PENETRANT INSPECTION
HT	HYD LEAK TEST
RT	RADIOGRAPHIC INSPECTION
UT	ULTRASONIC INSPECTION
* (REF)	INSPECT ROOT & FINAL PASS
(F)	INSPECT FINAL PASS ONLY

Drawing 50. Fuel Storage
 Tank Assembly
 (L159-HP005)
 NAA-SR-9213
 271

Drawing 51. Fuel Storage
Outer Liner
(L159-HP006)
NAA-SR-9213
273

Drawing 52. Fuel Storage
Inner Liner
(L159-HP007)
NAA-SR-9213
275

Drawing 53. Fuel Storage Rack
(L159-HP008)
NAA-SR-9213
277

NOTES: UNLESS OTHERWISE SPECIFIED
1. FOR GENERAL NOTES SEE AT DWG L159-HP008

Drawing 54. Fuel Storage
Rack Details
(L159-HP013)
NAA-SR-9213
279
681 IV.000019148

SECTION 'B-B'

- NOTES UNLESS OTHERWISE SPECIFIED
1. FABRICATE PER A1 SPEC
 2. ALL WELDS TO COMPLY WITH SECTION VIII OF THE ASME BOILER & PRESSURE VESSEL CODE
 3. ALL UNMACHINED SURFACES TO HAVE A SURFACE ROUGHNESS OF 250/ RMS MAX
ALL MACHINED SURFACES TO HAVE A SURFACE ROUGHNESS OF 125/ RMS MAX
 4. MACHINING TOLERANCES TO BE -
FRACTIONS - $\pm \frac{1}{32}$
DECIMALS - ± 0.010
ANGLES - $\pm \frac{1}{2}^\circ$

NOTES UNLESS OTHERWISE SPECIFIED
FOR GENERAL NOTES SEE AT DWG
L159 4P009

WELD	TEST SYMBOLS
SYMBOL	TEST
P.T.	PENETRATION INSPECTION
H.T.	HE LEAK TEST
R.T.	RADIOGRAPHIC INSPECTION
U.T.	ULTRASONIC INSPECTION
* (R.F.)	INSPECT BOTH & FINAL PASS
(F.)	INSPECT FINAL PASS ONLY

Drawing 57. Fuel Storage Top
Shield Details
(L159-HP010)
NAA-SR-9213
285

NOTES: UNLESS OTHERWISE SPECIFIED
1. FOR GENERAL NOTES SEE A.1.
DRAWING L159-HP009

Drawing 58. Fuel Storage Top
Shield Details
(L159-HP011)
NAA-SR-9213
287

CONTROL ROD STORAGE SLEEVE

- 1 HOT DIP GALVANIZE ALL SURFACES EXCEPT MACHINED SURFACES PER ASTM A123-53
- 2 MACHINED SURFACES TO BE PROTECTED BY A THICK COATING OF GREASE

Drawing 59. Control Rod Storage Cell
(L159-HP001)

Drawing 61. Process Flow
Diagram
(L159-SA010)
NAA-SR-9213
293
OSTI ID: 1000019148

- NOTES, UNLESS OTHERWISE SPECIFIED
- 1- WELD PER SECT VIII OF ASME BOILER AND PRESSURE VESSEL CODE.
 - 2- HELIUM LEAK CHECK PER A I SPEC NA 015-006
 - 3- ENDS OF TUBES (-5) TO HAVE A 1/8" FINISH WITH NO CHAMFERS.
 - 4- MATERIAL OF TUBES TO COMPLY WITH A I SPECIFICATION NB 0160-009 TYPE 1, CLASS 1.
 - 5- THE METHOD OF SECURING THE SPACERS (-4) TO THE TIE RODS (-9) SHALL BE APPROVED BY A I.
 - 6- MATERIAL OF SHELL TO BE 2 1/4" IN PER SPEC A 305-BST GRADE PEE.

ITEM NO.	NUMBER REQ.
X-10	3
X-25	26

THIS DRAWING IS NOT FOR CONSTRUCTION PURPOSES

ITEM NO	DIM-A	DIM-B	DIM-C	DIM-D	DIM-E
X-10	40'-4 13/16"	45'-10 1/16"	34'-3"	43'-3 5/16"	15 EQUAL SPACES @ 30 3/16" C.C. = 38 FT
X-25	44'-8 3/8"	50'-2 7/16"	38'-7 5/32"	47'-7 3/8"	15 EQUAL SPACES @ 33 7/8" C.C. = 42-3/8"

2	-11 LINER	8 1/2" OD x 1/4" W x 1" LG	SEE NOTE 6
2	-10 CROSS	10" x 8" SCH 40	SEE NOTE 6
8	-9 SPACER TIE ROD	7/16" DIA x 38" LG	1
1	-8 SPLIT LOWER SHELL	8" SCH 10 PIPE x 30" LG	1
1	-7 LOWER SHELL	8" SCH 10 PIPE x 30" LG	1
1	-6 SPLIT UPPER SHELL	8" SCH 10 PIPE x 30" LG	SEE NOTE 6
61	-5 TUBE	459 10 x 0.03 W x 39" LG	SEE NOTE 6
10	-4 L159-PE 00 5		
2	-3 L159-PE 00 1-2		
2	-2 L159-PE 00 4-1		
1	-1 ASSEMBLY		

Drawing 62. Low Temperature
Module Assembly
(L159-PE003)
NAA-SR-9213
295
OST 110 0000 1948

- NOTES UNLESS OTHERWISE SPECIFIED
- 1 WELD PER SECT XIII OF ASME BOILER AND PRESSURE VESSEL CODE
 - 2 HELIUM LEAK CHECK PER A.I. SPEC NA 0115-006
 - 3 ENDS OF TUBES (-5) TO HAVE A $\frac{1}{8}$ " FINISH WITH NO CHAMFERS.
 - 4 MATERIAL OF TUBES TO COMPLY WITH A.I. SPECIFICATION NB 0160 009 EXCEPT TYPE 321 S.S.
 - 5 THE METHOD OF SECURING THE SPACERS (-4) TO THE TIE RODS (-9) SHALL BE APPROVED BY A.I.

ITEM NO	NUMBER REQD
X 11	2
X 26	16

THIS DRAWING IS NOT FOR CONSTRUCTION PURPOSES.

ITEM NO.	DIM A	DIM B	DIM C	DIAMETER D
X-11	45'-6"	50'-11 1/2"	38'-5"	17 EQUAL SPACES @ 30" C C = 42'-6"
X-26	52'-2 3/8"	57'-7 7/8"	45'-4 3/8"	20 EQUAL SPACES @ 29 5/2" C C = 49'-2 1/2"

2	11	LINER	5/8" DIA. 1/4" W x 1" LG	304 S S	
2	10	CROSS	10 x 8 SCH 40	304 S S	
6	9	SPACER TIE ROD	3/4" DIA. x 45" LG	304 S S	
1	8	SPLIT LOWER SHELL	8" SCH 10 PIPE x 8" LG	304 S S	
1	7	LOWER SHELL	8" SCH 10 PIPE x 8" LG	304 S S	
1	6	SPLIT UPPER SHELL	8" SCH 10 PIPE x 8" LG	304 S S	
61	5	TUBE	435 10 x 0.085 W x 44" LG	SEE NOTE 4	
10	4	L159 PE005			
2	3	L159 PE008-2			
2	2	L159 PE002-1			
1	1	ASSEMBLY			
REF'D	PLANCH VIEW	PART NO.	DESCRIPTION	MATERIAL	QTY
LIST OF MATERIAL					

Drawing 63. High Temperature Module Assembly (L159-PE001)

NAA-SR-9213

297

OST ID: 000019M8

Drawing 64. Reheater Module
Assembly
(L159-PE006)
NAA-SR-9213
299
OSTID:000019148

Drawing 65. Steam Generator
Arrangement Plan
(L159-PE007)
NAA-SR-9213
301

Drawing 66. Steam Generator
Arrangement Section
(L159-PE008)

NAA-SR-9213

Drawing 67. Heat Balance Diagram
(L159-SA011)
NAA-SR-9213
305
OST ID: 000019148

Drawing 69. Nuclear Instru-
mentation System Block
Diagram
(L159-CA001)
NAA-SR-9213
309

REACTOR HEATERS		
REACTOR	HEATERS	WATERS
IN CORE, UPPER PLENUM	7	34.3
CALANDRIA	49	63.4
LOWER PLENUM	6	3.1
ACTIVE BOTTOM HEATERS	10	25.0
SPACE BOTTOM HEATERS	5	12.5

THERMOCOUPLE TABULATION		
SYMBOL	THERMOCOUPLE LOCATION	WATERS
TLP	LOADING FACE SHIELD	14
TCL	CAVITY LINER & BIOLOGICAL SHIELD	45
TS6	SOURCE SODIUM COOLANT	16
TTI	THERMAL INSULATION	4
TFE	FUEL ELEMENT COOLANT OUTLET	400
TTB	THERMAL SHIELD SURFACE	4
TCE	CONTROL ROD COOLANT OUTLET	48
TVW	VESSSEL WALL	14
TVL	VESSSEL LINER	14
TCW	CALANDRIA & LOWER PLENUM WALL	28
TIN	INLET NOZZLE	3
TIG	INLET GUARD PIPE	8
TVS	VESSSEL SUPPORTS	8
TCO	CALANDRIA HELIUM OUTLET	4
TCI	CALANDRIA HELIUM INLET	4
TVO	VESSSEL ANNULUS OUTLET	4
TOH	OUTLET NOZZLE	3
TOG	OUTLET GUARD PIPE	4
TNI	NUCLEAR INSTRUMENT THIMBLES	10

NUCLEAR INSTRUMENTATION		
W	CHANNEL	TYPE
1,2	SOURCE RANGE	PERIOD
3,4	INTERMEDIATE RANGE	COMP ION
5,6	POWER RANGE	ION
7,8	POWER RANGE	ION
9,10	POWER RANGE, LINEAR	COMP ION
11,12	FUEL IN CORE, FUEL LOADING, LOW POWER TEST	PERIOD

SCHEDULE FOR CORE LOADING *		
DESCRIPTION	REACTOR	CONTROL
CONTROL RODS	24	C
FUEL ELEMENTS	144	F
SOURCE, INITIAL LOADING	1	SI
SOURCE, NORMAL OPERATION	2	SO
POSITION CHAMBER	2	N

Drawing 70. Nuclear Instrumentation and Services Diagram (L159-CA005)
NAA-SR-9213
311
OSTI ID: 60001948

REACTOR & SODIUM LOOP THERMAL TRANSIENT PROTECTIVE INTERLOCKS.

1. PUMP COUPLINGS DE-ENERGIZED PRIOR TO MOTOR START UP.
2. THOSE CIRCUITS MARKED * HAVE CUTOFF ON LOW POWER OR FLOW.

Drawing 71. Plant Protective System Block Diagram (L159-CA002)

REFERENCE
 REACTOR INSTRUMENTATION
 & SERVICES DIAG L159-CA005

THERMOCOUPLE INPUTS		
TYPE	No INPUTS	No LOGGED
CR/C TYPE E	600	250
C/A TYPE K	150	50
TOTAL	750	300

Drawing 72. Data Logger/
 Computer Diagram
 (L159-CA004)

INSTRUMENT AIR COMPRESSORS

K-7A & K-7B (SPARE)
CAPACITY: 150 SCFM (EACH)
TEMP: 100 °F
PRESS: 125 PSIG (EACH)

INSTRUMENT AIR MOISTURE SEPARATOR

T-31A & T-31B (SPARE)
(SUPPLIED WITH COMPRESSOR)

INSTRUMENT AIR RECEIVER TANK

T-33
CAPACITY: 35 FT³
TEMP: 300 °F
PRESS: 150 PSIG

INSTRUMENT AIR PREFILTER

40 MICRON
(SUPPLIED WITH DRYERS)

INSTRUMENT AIR DRYERS

D-1A & D-1B (SPARE)
DESSICANT TYPE

INSTRUMENT AIR AFTERFILTER

10 MICRON
(SUPPLIED WITH DRYERS)

NOTE:
1. * DENOTES SUPPLIED WITH COMPRESSOR.
2. FOR LEGEND & SYMBOLS SEE DWG. L159-SA001

PLANT AIR COMPRESSOR

K-5
CAPACITY: 150 SCFM
TEMP: 100 °F
PRESS: 125 PSIG

PLANT AIR MOISTURE SEPARATOR

T-32
(SUPPLIED WITH COMPRESSOR)

PLANT AIR RECEIVER TANK

T-34
CAPACITY: 19 FT³
TEMP: 300 °F
PRESS: 150 PSIG

Drawing 74. P & I Diagram
Plant and Instrument
Air Systems
(L159-SA006)

NAA-SR-9213

319

OSTI ID: 6000 19148

Drawing 75. P & I Diagram
Helium Supply System
(L159-SA002)

NAA-SR-9213
321

OSTI ID: 060019148

T-20
(LEASED EQUIPMENT)

Drawing 76. P & I Diagram
Nitrogen System
(L159-SA003)
NAA-SR-9213
323
OSTI ID: OOOO 1948

REACTOR CAVITY COOLERS
X-22 a & b

CELL ATMOSPHERE COOLERS
X-13 a & b

NOTE: FOR LEGEND SEE DWG. L159-SA001

Drawing 77. P & I Diagram
Cell N₂ Cooling System
(L159-SA008)

Drawing 78. P & I Diagram
R/A Vent System
(L159-SA004)
NAA-SR-9213
327
8571 IP 0000 19148

Drawing 79. P & I Diagram
R/A Liquid Waste System
(L159-SA005)
NAA-SR-9213
329
OSTI ID: 000019148

SECTION 13-13

SECTION A-A

ELEVATION

Drawing 80. Maintenance
Cell
(L159-HP014)
NAA-SR-9213
331

THIS CELL USED FOR WASHING FOLLOWING

- 1 SODIUM PUMPS
- 2 FUEL HANDLING CASK INTERNALS
- 3- HANDLING TOOLS
- 4 MISC IRRADIATED ASSEMBLIES REMOVED FROM THE REACTOR

- 1 HOT DIP GALVANIZE LINER SURFACES EXCEPT MACHINED SURFACES PER ASTM-A123-53
- 2 CONTAINERS ARE CARBON STEEL-COATED WITH A CAUSTIC AND RADIANT RESISTANT PAINT

Drawing 81. Component
Wash Cell
(L159-HP002)
NAA-SR-9213
333

Drawing 82. P & I Diagram
Steam Dump and Startup
System
(L159-SA007)
NAA-SR-9213
335
OSTI ID: 600019148