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Introduction

One of the fundamental reasons for developing methods of power generation utilizing nuclear fission as the heat source is to extend the world's supply of energy resources beyond those available in the conventional fuels; coal, oil and gas. The only naturally occurring isotope which can sustain a chain reaction is U-235. In the fertile materials U-238 and Th-232, there is a possibility of magnifying the supply of nuclear energy by several orders of magnitude.

In certain types of reactors, it is possible to produce more fissionable material than is consumed in the production of heat. This phenomenon is called breeding and is the main reason for the effort to develop fast reactors.

It is the purpose of this paper to discuss some of the distinguishing characteristics of a fast reactor using as illustration the reactor being developed by Atomic Power Development Associates, Inc. (APDA) for contruction by Power Reactor Development Corporation (PRDC) at the Enrico Fermi Atomic Power Plant near Monroe, Michigan.

General

The fate of a neutron in a reactor is to leak out, be absorbed or cause fission. In a fast reactor, the approach is to reduce parasitic absorptions so that more neutrons are available to produce fissionable material from fertile. There is an added attraction in the neutron balance with a high energy spectrum in that U-238 fission adds to the neutron population and thus helps the breeding ratio.

Three nuclear parameters of interest in attempting to evaluate the breeding possibilities of a particular fuel are:

v—number of neutrons emitted per fission
η—number of neutrons emitted per fuel absorption
α—ratio of radiative capture cross section to fission cross section for fuel

Thus

$$\eta = \frac{\sigma_f}{\sigma_a} v = \frac{\sigma_f}{\sigma_f + \sigma_c} v = \frac{v}{1 + \sigma_c/\sigma_f} = \frac{v}{1 + \alpha}$$

Since one neutron is needed from each fission to sustain the chain reaction, the maximum breeding ratio is $\eta - 1$. Early experiments indicated that η is insensitive to the energy of the incident neutron, therefore, an improvement in α with neutron energy would increase breeding possibilities. Table 1 shows a comparison of these parameters for fast and thermal reactors.

Table 1
Comparison of Nuclear Parameters
for Fast and Thermal Reactors

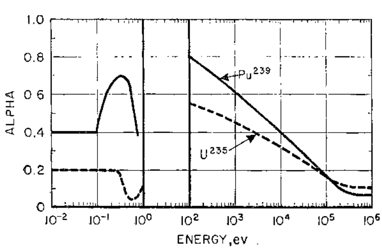
	ν		η		α	
	Fast 0.2 mev	Thermal 0.025 ev	Fast 0.2 mev	Thermal 0.025 ev	Fast 0.2 mev	Thermal 0.025 ev
U-235	2.46	2.46	2.20	2.08	0.11	0.184
U-233	2.54	2.54	2.49	2.31	0.02	0.1
Pu-239	2.88	2.88	2.64	2.03	0.09	0.42

From this table it can be seen that breeding looks more probable in a fast reactor than in a thermal, with plutonium being the outstanding choice for a fuel. Plutonium fueled power reactors cannot be built at the present time principaly because the metallurgy of plutonium is not well enough known. At today's stage of development, U-235 is being used as the fuel for fast power breeders. Thermal breeding with U-233 looks feasible, and will not be considered in this discussion although it is not at all excluded from being a possibility with fast neutrons.

Figure 1 shows the variation of α with energy. It will be noted that for neutron energies above 0.1 mev there is a significant improvement in the value of α over the thermal value. Again the superiority of Pu-239 over U-235 is noted.

It is essential to maintain the neutron spectrum at a high energy to capitalize on this low value of α and this can be accomplished in a low power reactor. However, in a high power reactor, it is necessary to dilute the U-235 with U-238 in order to get enough heat transfer surface to cool the core. The spectrum is thereby lowered by the inelastic scattering of the U-238, and the value of α is raised so that the situation is not as attractive.

Breeding is still practical in a high power reactor despite the



VARIATION OF ALPHA WITH ENERGY
Fig. 1

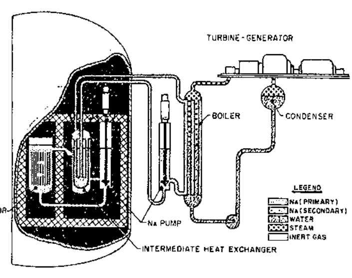


DIAGRAM OF NUCLEAR POWER PLANT
Fig. 2

higher $\propto$ because of the fast fissions in U-238 and the low absorption cross-section of structural materials and coolant. Table 2 is a neutron balance for the APDA reactor.

Table 2

	Neutron Production Core	Neutron Production Blanket	Neutron Production Total	Neutron Absorption Core	Neutron Absorption Blanket	Neutron Absorption Total
U-235 fission	0.841	0.024	0.865	0.339	0.010	0.349
U-238 fission	0.085	0.050	0.135	0.034	0.020	0.054
U-238 capture				0.141	0.353	0.494
U-235 capture				0.076	0.003	0.079
*Zr capture				0.003		0.003
*SS capture				0.004	0.003	0.007
*Na capture				0.001	0.001	0.002
*Mo capture				0.002		0.002
*B_4C capture				0.004		0.004
TOTAL	0.926	0.074	1.000	0.604	0.390	0.994
Neutron Leakage: Core to blanket				—0.322		
Blanket to shield				—0.006		

*The functions of these materials will be discussed later in the paper.

The number of plutonium atoms produced per atom of U-235 destroyed or the breeding ratio can be approximated by:

$$BR = \frac{\text{U-238 captures}}{(\text{U-235 fissions} + \text{U-235 captures})\ \text{core}} = \frac{0.494}{0.339 + 0.076} = 1.19$$

Some Characteristics of Fast Reactors

The design of a fast reactor has some inherent distinguishing characteristics which lead to some particular problems which must be recognized and solved.

1. High Critical Mass

The critical mass of a fast reactor is higher than the critical mass of a thermal one since the fission cross-section of the fuel is lower. The mere fact that so much uranium is present leads one to consider the possibilities of a supercritical mass forming if some accident causes the fuel to melt. A good deal of effort is being put forth to determine analytically and experimentally what a meltdown of the fuel in the Enricho Fermi reactor would look like and how it would behave in the liquid state. Argonne National Laboratory, where much of this country's fast reactor work is being done, is designing a test reactor of about 100 kw steady-state power, but capable of absorbing in its own heat capacity about 200 mw-sec. of energy from a transient. In this facility, some fuel elements can be put into test holes and actually melted to trace the flow of the melted fuel. There will be an area incorporated in the bottom of the Fermi reactor vessel to insure that the molten fuel running down there will be in a subcritical geometry. Of course, every conceivable practical safeguard against a possible meltdown situation is being designed into the plant so that the necessity of relying on safe meltdown is remote.

2. **Small Core**

The core of a fast reactor is small since it is essentially uranium and coolant. To remove large amounts of heat from a small core, the fuel must be finely subdivided to limit the temperature gradient across it. In a heterogeneous reactor this leads to quite small fuel elements or even smaller elements grouped into bundles with flow passages within the bundles. The close tolerances in fabrication and assemby required tend to increase fuel element cost.

3. **Coefficients of Reactivity**

Thermal reactors usually have a large negative temperature coefficient of short time constant associated with the moderator. Fast reactors do not have this characteristic of one dominant fast-acting coefficient. Therefore, care must be taken in design to be certain of reactor stability and safety based on a number of small coefficients of various time constants. Analysis of the reactor's dynamic behavior is thus complicated by a number of feedbacks of different time constants.

One of the temperature effects which has been of particular interest in the design of fast reactors is the Doppler coefficient. When a particular cross-section contains resonances at certain incident particle energies, the shape of the resonance depends upon the relative speed between the incident particle and the target nucleus. Thus, for a given neutron energy, the fission and absorption rates depend upon the thermal agitation of the surrounding material.

A good deal of theoretical and experimental work has been done by people concerned with fast reactors on the sign and magnitude of this Doppler effect. They have been more concerned with this than present designers of thermal reactors for two reasons. First, all the temperature coefficients are small so that any effect may be a major influence in the net result. Second, the cross-section resonances are not as well known or as easily determined at higher energies. In order to be certain that the Doppler coefficient would not be positive in the Fermi reactor, a minimum U-238 to U-235 ratio was calculated and set at 2 to 1. Based on some recent experimental work by Argonne National Laboratory,[1] Bethe[2] has calculated that a 1 to 1 ratio is sufficient. So the Doppler effect does not appear to be as troublesome as originally anticipated.

The core of a fast reactor is small, and therefore small geometry changes have appreciable effects on reactivity. In a reactor where the neutron flux decreases as distance from the core center increases, a given fuel element will be hotter on the side closest to the core centerline. Thus, as temperature is increased, the fuel element will tend to bow. It is an important consideration in fast reactor design to make certain that this bowing is either eliminated or made to be in such a direction as to decrease reactivity as temperature is increased.

4. Small Amount of Excess Reactivity Needed

The excess reactivity which must be built into a fast reactor is smaller than a thermal one because:

a. Small negative temperature coefficient

b. Poisoning effect of fission products is small

Table 3 shows a breakdown of the excess reactivity in the Fermi Reactor.

Table 3
Reactivity in Shim Rods

Unit of Dollar* Based on Delay Fraction, β, of 0.00741		
Temperature override	0.20	dollars
Burn-up weekly unloading	0.33	"
Fuel element growth	0.07	"
Fission product poisoning	0.02	"
Control margin	0.30	"
Total excess (Shim rods)	0.92	"

* One dollar is the amount of reactivity required to make a reactor prompt critical.

5. Neutron Lifetime

The prompt neutron lifetime in a fast reactor is about 10^{-7} sec, while the prompt lifetime in a thermal reactor may be as long as 10^{-3} sec. This lifetime, however, is important only when prompt critical is approached so that in the normal situation of delay critical, thermal and fast reactors behave in very similar manners. As can be seen from Table 3, above, the available reactivity in a fast reactor is in the order of one dollar so it is a remote possibility that the reactor could even be prompt critical.

Although the delayed neutron fraction for the fast fission of U-235 is smaller than for thermal fission, the fast fission of the U-238 diluent introduces its higher delay fraction. The net result is that the delay fractions for fast and thermal reactors are about the same.

6. Shielding

Shield design for fast reactors is different from the design for thermal reactors because no moderating material can be placed close to the core. The neutron spectrum which must be shielded against is, therefore, of intermediate energy. The shield will tend to be slightly thicker than for a thermal reactor, since the neutron energy is too low for inelastic scattering to be effective, but high enough so that many elastic scatters are required to insure absorption. However, since the fast reactor is small, the total volume of shielding for a fast reactor will be somewhat less than for thermal.

7. **Burn-up**

Fuel element life in a fast reactor is limited by radiation damage rather than the fission product poisoning limit encountered in many thermal reactors. The atoms destroyed in a fast reactor are, therefore, usually quoted as a percentage of the fuel alloy. Thermal reactor burn-up is usually quoted as a percentage of uranium alone.

Since the critical mass in a fast reactor is high, the cost of a large hold-up of material in the whole fuel cycle is large. High burn-up is thus an important stepping stone toward economical power. The limit of burn-up based on radiation damage is a metallurgical problem rather than some fundamental problem associated with the reactor so that future experimentation and study is certain to extend the possible irradiation of fuel elements.

Enrico Fermi Plant

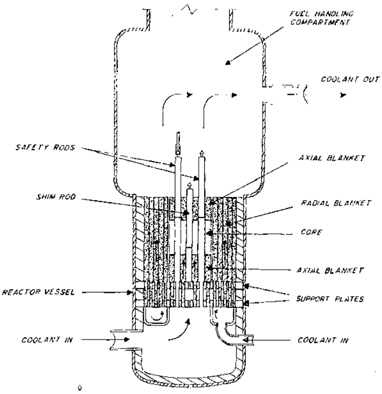
The Enrico Fermi Atomic Power Plant is being built by Power Reactor Development Company (PRDC) from a conceptual design furnished by Atomic Power Development Associates, Inc. (APDA). Both of these companies are non-profit, membership organizations including 53 companies from 17 states. The reactor is being constructed as part of the AEC Power Reactor Demonstration Program, and is intended to supply necessary information on the performance and operation of a commercial-size fast breeder system. The plant is scheduled for operation in early 1960.

PRDC will own and operate the reactor plant and will sell steam to the Detroit Edison Company who will construct a turbine generator unit at the plant site.

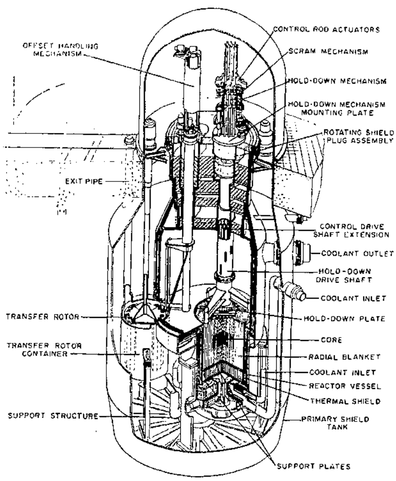
Reactor Plant

The reactor will be a fast breeder utilizing U-235 as the fuel and U-238 as the fertile material for the production of Pu-239. Reactor power will be 300 mw of heat and the electric plant will have an output of 100 mw.

A schematic diagram of the plant is shown in Figure 2. Liquid sodium is used as the reactor coolant and is pumped into the bottom of the reactor vessel at 550° F. The sodium is heated to 800° F. and flows to the intermediate heat exchanger where it transfers its



REACTOR ARRANGEMENT
Fig. 3



PERSPECTIVE VIEW OF REACTOR
Fig. 4

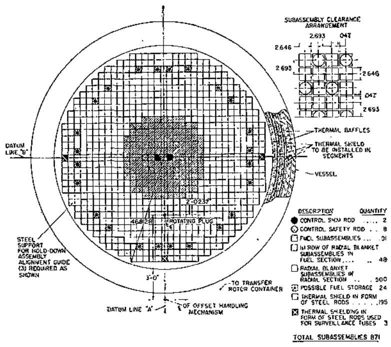
heat to a secondary loop of sodium. The secondary loop serves to isolate the radioactive sodium from the water in the boiler and also allows the radioactivity to be contained within shielding. The steam conditions to the turbine are 730° F., 600 psi.

Figure 3 is a schematic of the reactor and indicates the core which is approximately a right cylinder 30 inches in diameter and 30 inches high. The average effective flux in the core is 5×10^{15} n/cm^2-sec having an average neutron energy of 0.25 mev. The critical mass is 485 kg of U-235.

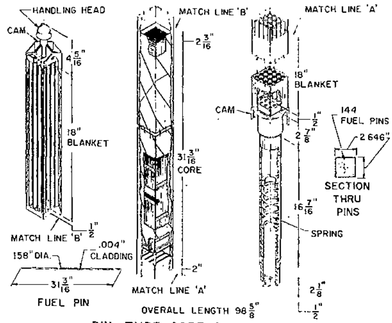
There are eight safety rods and two shim rods containing boron carbide as a poison material for control and hold-down reactivity. Each safety rod contains one dollar of poison for a total hold-down reactivity of $8.00. As mentioned previously, the two shim rods have a combined worth of 92 cents.

Figure 4, Perspective View of Reactor, shows the devices which allow the remote removal and introduction of core and blanket subassemblies. The combined eccentric rotation of the offset handling mechanism and the rotating shield plug enable the pick-up device to be located over any desired subassembly. For removal, the subassembly is deposited into the transfer rotor and rotated to be taken out through the exit pipe and into a shielded cask car. The introduction of a subassembly is accomplished by a reverse procedure.

A hold-down mechanism and hold-down plate are also shown. Their function is to clamp the core subassemblies in place against the inlet sodium pressure of 100 psi. The radial blanket subassem-



REACTOR CROSS SECTION
Fig. 5



PIN-TYPE CORE SUBASSEMBLY
Fig. 6

blies are fed from a lower pressure plenum and do not require any hold-down.

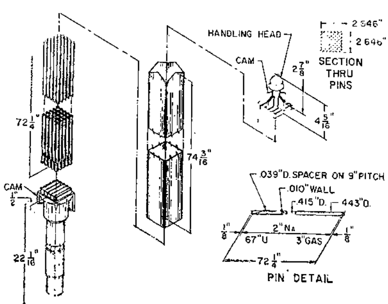
Figure 5, Reactor Cross-Section, shows the arrangement of the core and blanket subassemblies, shim and safety rods, thermal shielding and temporary storage space for 26 spent core subassemblies.

Core and Blanket Subassemblies

The core subassemblies contain depleted uranium rods in stainless steel cans to form the upper and lower axial blankets as shown in Figure 6. The core section is located in the central portion of the subassembly and consists of 144 zirconium clad pins of a uranium—10 wt% molybdenum alloy. The uranium is enriched to 27% in U-235. The lower ends of these pins are anchored but the upper ends are free to allow for expansion. There are 91 of these subassemblies in the reactor.

The cam indicated at the bottom of the square portion of the subassembly is a self-orienting device to make certain that the subassembly can be lowered into its square hole formed by the four adjacent subassemblies.

Figure 7 shows a radial blanket subassembly which contains rods of depleted uranium sodium bonded to stainless steel cans.



RADIAL BLANKET SUBASSEMBLY
Fig. 7

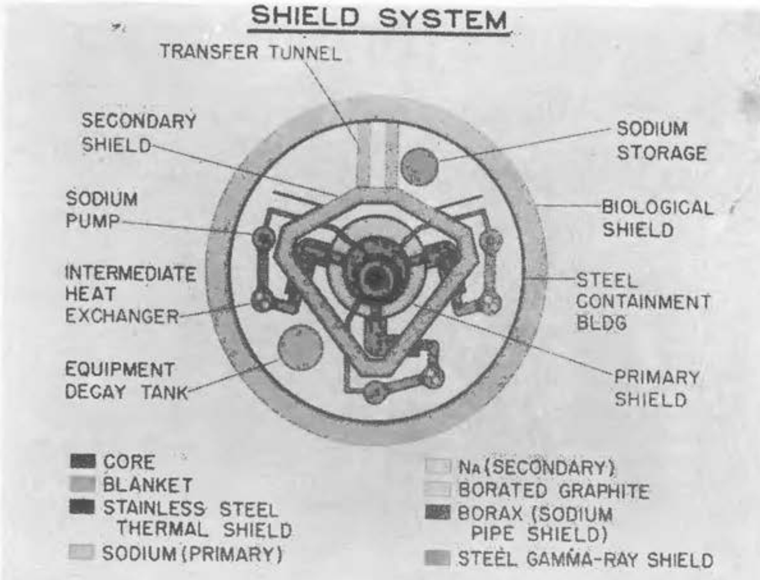


Fig. 8

Shield

The shield system for the Enrico Fermi reactor shown in Figure 8, is designed to limit plant personnel dose to 1/10 of the total radiation dose presently allowed in AEC installations during a 40-hour work week. The resulting dose limit is 30 milliroentgens per week. Additional criteria for the shield system design are:

a. Equipment in the secondary sodium loops will not be allowed to become radioactive from neutron bombardment. The neutron flux surrounding this equipment will be reduced to 10^4 n/cm^2-sec or lower. This flux is below the significant activation level for the steel and sodium.
b. The total exposure of the reactor vessel over the life of the plant will be kept within present experimental knowledge of 10^{22} nvt. To accomplish this, the neutron flux is held below 3×10^{13} n/cm^2-sec. at the vessel.
c. The ambient temperature of concrete will be kept below 200° F in order to prevent loss of water. The total energy flux of neutrons and gamma rays allowable is $4 \times {}^{10}$ mev/cm^2-sec.

Site and Building

The reactor and associated steam plant will be built on a site of approximately 915 acres exclusion area. Figure 9 indicates its location on the western shore of Lake Erie aobut 30 miles southwest of Detroit and 25 miles northeast of Toledo.

Fig. 9

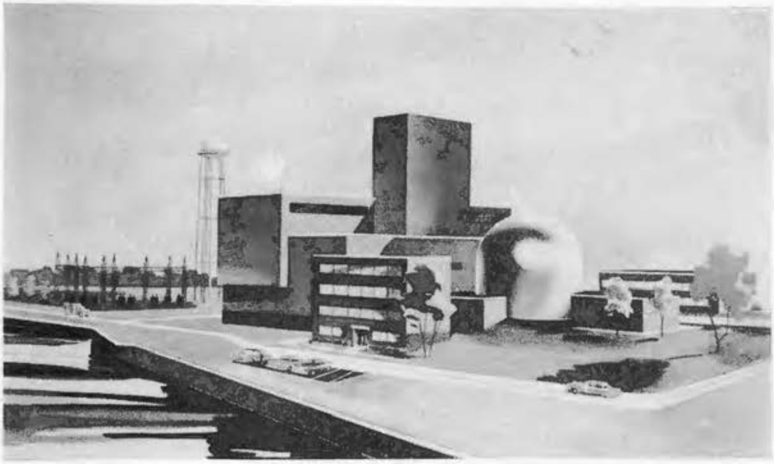


Figure 10 is an artist's concept of the final plant and shows the steel-domed reactor building. The building is 72 ft. in diameter, has an overall height of about 120 ft. It is set about half below ground. It is designed to contain all radioactivity in the event of an incident and can withstand an internal pressure of 32 psig at 650° F.

APPENDIX

Enrico Fermi Atomic Power Plant
Reactor Design and Performance Data Tabulation[3]

Core Design and Performance		
Core power	—kw	268,000
Core volume	—cu ft	11.65
Average heat flux	—Btu/hr-sq ft	665,000
Maximum to average heat flux	—	1.43
Power density	—kw/cu ft.	23,000
Specific power	—kw/kg U-235	553
Diameter	—in.	30.5
Length	—in.	31.2
Core sodium flow rate	—lb/hr	11,880,000
Sodium velocity	—ft/sec	32.5
Core comosition	—% volume	
U-235		7.75
U238		21.35
Zr, Mo. SS		24.7
Na		46.2
Number of fuel subassemblies	—	91
Number of fuel pins per subassembly	—	144
Fuel alloy—OD	—in.	0.150
Cladding—OD	—in.	0.158
Cladding Thickness	—in.	0.004
Maximum fuel temperature	—F	1,300
Total fuel alloy burn-up	—atomic %	1.0
Blanket Design and Performance		
Blanket power	—kw	
Radial section		28,000
Axial section		4,000
Blanket volume	—cu ft	
Radial section		161
Axial section		13

Average heat flux	—Btu/hr sq ft	
Radial section		8,500
Axial section		13,000
Maximum to average heat flux (radial)	—	55
Radial blanket dimensions	—in.	
Inside diameter		30.5
Outside diameter		78.5
Length		67
Axial blanket dimension	—in.	
Diameter		30.5
Length		18
Reactor Control		
Safety elements		
Number	—	8
Reactivity per rod	—dollars	1.00
Total stroke	—in	54
Length of poison material	—in.	33
Maximum run-in of safety rods	—dollars/sec.	0.75
Scram of safety rods	—dollars/sec.	25
Boron-10 required, per rod	—kg	0.90
Shim control rods	—	
Number	—	2
Total reactivity in both rods	—dollars	0.92
Total stroke	—in.	15
Maximum reactivity insertion rate	—dollars/sec.	0.01
Boron-10 required, per rod	—kg	0.47
Reactor Physics		
Critical mass U-235	—kg	485
Core conversion ratio	—	0.34
Blanket conversion ratio	—	0.86
Total conversion ratio	—	1.20
Radial/axial blanket Pu production	—	6
Average core neutron energy	—mev	0.25
Average effective core flux	—n-cm^2/sec.	0.5×10^{16}
Average generation time	—sec.	0.09
Delayed neutron fraction	—	0.00741
Mean delay time of delayed neutrons	—sec.	12
Prompt neutron lifetime	—sec.	2×10^{-7}
Temperature coefficients of reactivity	—Δk/k/c	
Core fuel expansion	—	-2.5×10^{-6}
Ejection of compressed sodium	—	-0.6×10^{-6}
Sodium heating	—	-7.1×10^{-6}
Top support expansion	—	-3.0×10^{-6}
Bottom support expansion	—	-3.0×10^{-6}
Doppler effect	—	-3.1×10^{-6}
Total Core	—	-19.3×10^{-6}
Blanket coefficients of reactivity	—Δk/k/c	
Uranium expansion	—	-0.8×10^{-6}
Sodium heating	—	-3.3×10^{-6}
Structure expansion	—	-0.6×10^{-6}
Total Blanket	—	-4.7×10^{-6}
Total reactor coefficient of reactivity	—Δk/k/c	-24×10^{-6}
Liquid Metals and Steam System		
Gross electric capacity	—mw	100
Net electric power output	—mw	90
Net thermal efficiency	—%	30.0
Sodium temperatures	—F	
Leaving reactor	—	800
Entering reactor	—	550
Sodium flow	—lb/hr	13.2×10^{6}
Secondary sodium temperatures	—F	
Entering boiler	—	765
Leaving boiler	—	515

Secondary sodium flow	—lb/hr	13.2×10^6
Steam pressure	—psia	600
Steam temperature	—F	755
Feedwater temperature	—F	400
Steam flow	—lb/hr	1.02×10^6

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