MASTER

988
GEAP-5678

1000 MWe

L M F B R

Liquid Metal Fast Breeder Reactor

FOLLOW-ON STUDY

GENERAL ELECTRIC
TASK II REPORT

**Conceptual Plant Design, System
Descriptions, and Costs for a 1000
MWe Sodium-Cooled Fast Reactor**

Prepared for ARGONNE NATIONAL LABORATORY

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United States Atomic Energy Commission

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CONCEPTUAL PLANT DESIGN, SYSTEM DESCRIPTIONS, AND COSTS
FOR A

1000 MWe SODIUM COOLED FAST REACTOR

TASK II REPORT OF 1000 MWe LMFBR
FOLLOW-ON WORK

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ADVANCED PRODUCTS OPERATION
GENERAL ELECTRIC
SUNNYVALE, CALIFORNIA

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SECTION I INTRODUCTION

1.1 OBJECTIVES OF STUDY (A. S. Gibson)

This work was performed for Argonne National Laboratory as a part of a continuation of the 1000 MWe Liquid Metal Fast Breeder Reactor (LMFBR) design studies previously performed for the AEC. ^(1, 2) A major objective of this follow-on work is to provide a relative comparison of the cost incentives for various design alternates and to consider the cost incentives against the required development to achieve the state-of-technology implied. A further objective is to provide a reference design with sufficient detail to permit a better definition of absolute costs than was possible in previous studies. To implement these objectives, five design tasks were identified by Argonne National Laboratory; these are:

Task I

The development of one or more nuclear steam supply system concepts, with certain trade-off studies to aid in the definition of these concepts. The selection of a reference concept for further study.

Task II

Preparation of system design descriptions defining the plant systems and components for the reference concept, and a cost estimate of the reference concept.

Task III

Evaluation of trade-off and optimization studies in which numerous design parameters and economic ground-rules are varied to determine the effect on the economics of the reference concept.

Task IV

Preparation of an R&D program to accomplish the state-of technology implied by the previous three tasks and by Task V.

Task V

Selected safety studies.

The results of the Task I effort have been described in the Task I Report: "Comparison of Two Sodium-Cooled 1000 MWe Fast Reactor Concepts," GEAP 5618, June, 1968. ⁽³⁾

This document presents the results of the Task II effort. The reference concept is defined, system descriptions are provided, and capital costs, fuel cycle costs and operating and maintenance costs of the reference concept are defined.

1.2 GENERAL APPROACH (A. S. Gibson)

During the Task I effort, General Electric evaluated two different core design concepts solely on the basis of different safety criteria. In one concept, entitled the "Advanced Design," it was assumed that accidents leading to complete core voiding and major meltdown of the fuel either could be prevented by sophisticated accident detection and protection devices, or could be shown to be "incredible" (of acceptably low probability) by further development effort. In the second concept, entitled the "Conservative Design," it was assumed that major voiding and meltdown accidents could occur, but the severity of such accidents would be minimized by designing particular characteristics into the reactor core (e.g. enhanced doppler by BeO and reduced

void reactivity by pancaking). Based upon the results of the evaluation of these two concepts, the advanced design was selected as the reference concept for further evaluation.

Following the selection of the reference concept, numerous economic optimization and trade-off studies were performed by varying key design parameters or characteristics of the reference concept and determining the effect of this parametric variation on economics. These studies included an evaluation of alternate fuel design parameters, different fuel types (vented versus non-vented and oxide versus carbide), alternate core geometries, different sodium and steam temperatures, alternate steam and sodium cycle arrangements, different component arrangements, variation of economic criteria, etc. This optimization effort constituted the Task III work and will be reported in the Task III report.

The optimization and economic trade-off work for Task III made possible the selection of economically desirable reactor and system parameters and characteristics for the reference concept. The design parameters and characteristics were optimized within the limits permitted by the contract, by the uncertain technological state, and by the uncertain economic predictions for the 1980's. Based upon these selected parameters and characteristics, system descriptions, and conceptual specifications were prepared; equipment arrangement and mechanical design drawings were generated; and cost estimates of the reference plant were made. The description of this plant, its systems and components and the associated costs are reported herein. This Task II material is thus a description of what constitutes the reference concept. The Task III report will be a description of why the reference parameters were selected, and how the work was done.

1.3 LIMITATIONS OF STUDY AND FUTURE WORK (A. S. Gibson)

The validity of the Task II study effort is, to a large extent, dependent upon the methods used and the technological basis for the various selections made. As noted above, a description of the methods used and the reasons for the parametric selections will be given in the Task III report. However, brief mention should be made of some of the methods used for this Task II work.

In general, the computer programs and data were similar to those applied to the Task I work, and described in the Task I report with a few important differences. As noted in that report, the number of codes and data available permitted a degree of sophistication in the analysis which is usually not attempted in conceptual design studies. For example, most of the nuclear analysis work was performed using the BISYN two-dimensional diffusion theory synthesis code, ^(4, 5) and estimates of corrections to particular characteristics were applied based upon transport theory work. The nuclear cross sections used in performing these analyses were generated for appropriate neutron spectra corresponding to the core composition. The thermal-hydraulic analyses were based upon recent experimental data and included the inter-relationship between core pressure drop and structural steel required in the core (which in turn was reflected in the nuclear analyses). All of the analyses were evaluated against specified safety criteria.

In addition to the integration of disciplines in core design noted above, the Task III work involved extra effort in four major areas which decreased the uncertainty associated with the Task II design compared with the Task I work. These major areas of effort are:

- 1 Integration of system optimization with the core optimization;
- 2 Generation of core fuel costs based upon a fuel model which considers the interacting effects of fuel swelling, cladding allowable strain, fuel density, cladding temperature, and burnup;
- 3 Avoidance of arbitrarily fixing numerous design parameters; and,
4. Input of more realistic nuclear data to the core designs.

The integration of the core and system optimization studies permitted an optimization of such parameters as coolant outlet temperature. This was accomplished by manipulating the different for the core and the system. For the core, increased coolant outlet temperature resulted in increased fuel cost due to the decreased strength of the fuel cladding material. For the system, increased temperature resulted in decreased cost due to the increased cycle efficiency, despite decreased component material strength. A balance of these two effects resulted in the selection of optimum temperature.

In the generation of fuel costs, assumptions were made which relate fuel swelling to fuel density and to fuel burnup, and which relate allowable cladding strain to cladding temperature and to cladding fluence. The chosen relationships are based on the limited data which is currently available. The effect on fuel cost of alternate design criteria and economic criteria was also evaluated.

Numerous design parameters were arbitrarily fixed in Task I in order to minimize the labor associated with comparing the advanced and conservative concepts. For example, three key design parameters which were arbitrarily assigned values are: the gas plenum height to core height ratio (1.25), the

refueling interval (6 months), and the number of coolant loops (6). For Tasks II and III, studies were made to optimize these values or to select them based upon a more realistic appraisal of what they should be. The gas plenum height was sized for both vented and sealed pin designs with the reference for Task II being vented (based upon economic desirability and qualitative safety considerations). The refueling interval was lengthened to 12 months to minimize down time and to match utility preference. The number of coolant loops was set at three based upon economic and technological considerations. Other parameters fixed in Task I but allowed to vary in Task III included fuel cladding thickness, number of enrichment zones, type of plant cycle, steam-generator arrangement, and main steam conditions.

During Task I the nuclear analyses were performed utilizing the ENDF/B cross section file. As a result of recent British experiments, it appeared prudent to increase the values of the plutonium capture-to-fission ratio (α). More recent measurements by Oak Ridge National Laboratory on the Rennselaer Polytechnic Institute LINAC tended to confirm the need for increasing the plutonium α value. This was done for the majority of the Task II and Task III work. Moreover, nuclear business growth studies within General Electric demonstrated that in the 1980 decade and possibly in the 1990 decade, most of the plutonium feed material for fast reactors will be produced by water reactors. This plutonium feed is rich in plutonium 241 (compared with fast reactor discharge plutonium), and has the effect of improving breeding ratios, and reducing doubling time. Water reactor discharge plutonium was therefore used as the feed material for the majority of the Task II and Task III work.

Despite the additional effort in the Task II and Task III work compared with Task I, there are numerous limitations which stem from causes similar to those in Task I, namely:

- a. Narrowness of contract scope,
- b. Incompleteness, and
- c. Technological problems.

These limitations will be discussed in some detail in the Task III report, but some mention should be made of problems which bear directly on the Task II work.

The contract defined a conceptual design effort for Task II. The conceptual nature of the work precluded extensive investigations into various design areas, even though the technology was available to make such investigations. Two areas, for example, which could significantly alter some of the design features of the reference plant upon further study are the structural and system analysis of the primary sodium vessel and components, and the analysis of alternate plant control systems. A reference arrangement was selected for the Task II primary system vessel and components, and a reference plant control system is provided, but the limited scope of the present effort prevented sufficient analyses to determine detailed solutions to many of the design problems identified in this conceptual effort. The Task IV R&D report will identify these areas of weakness. Fortunately, the problems are usually of such a nature that the cost estimate of the plant should not be seriously affected by alternate design solutions which result from more detailed analysis. However, some of the design parameters or design arrangements could be significantly altered by such analysis.

Most of the limitations produced by incomplete analysis result in modest off-optimum conditions which do not seriously affect the economic results or major conclusions of the study, but do, in some cases, yield design parameters which would be modified if the study were carried further. These situations result not because of contractual scope limitations, but because the additional effort was

not deemed to be worthwhile when compared to the expected result. For example, the Task I work showed that for the assumptions and criteria used in that study, the advanced core had an optimum active height of 36 inches. At the commencement of the Task III effort it was decided to reduce this core height to 30 inches (at a slight economic penalty) in order to reduce pressure drop and thus simplify the development problems associated with the mechanical design of the core. Other work in Task III showed that use of vented fuel could significantly reduce the fuel plenum height (and core pressure drop), and more complete optimization of the fuel pin diameter resulted in a slight increase in diameter (to 0.25 inches o.d.). All of these effects combined to result in a core pressure drop reduction from 190 psi in Task I to 64 psi in Tasks II and III. More complete optimization with the core and system may have resulted in a slight increase in core active height to something between 30 and 36 inches, but the effort was not considered worthwhile. Obviously, the difference in core pressure drop of 190 psi and 64 psi is significant, and has implications regarding pump development and core structural development, but it is doubtful that additional analysis would have reduced development uncertainty in these particular areas.

The greatest cause of a limitation on the results of the Task II and Task III work is that of technological uncertainty. Several important examples can be cited. A major technological uncertainty in the reference concept involves the characteristics and types of instruments which are provided in the core for detection of local accident conditions. Further, it is not certain that any such instruments will be required. Numerous possible devices will be described in the Task III report with the requirements related to possible accident situations. However, the state of technology is still so early that a precise definition or "system description" with specifications of specific instruments required for

the safety system is not possible. Work conducted as a part of the Task III and Task V studies indicated that the development of suitable sensors, logic systems and protective designs by the 1980's has a high probability of success, but precise identification cannot now be made. Promising instrument sensors and logic systems are included herein as illustrative examples, and the costs are estimated as if the items were developed (with fairly large contingencies included).

Numerous technological uncertainties are associated with the design and development state of the components. This includes the reactor structure, the pumps, sodium to sodium heat exchangers and steam generators. In some cases particular parameters were arbitrarily limited to maximum or minimum values even though an economic incentive might have been shown to exceed a given value. For example, the selection of steam conditions of 2400 psia, 950°F/950°F represented an optimization up to and including the selected pressure. It appeared that further work might have shown a modest economic incentive for higher pressure up to and above supercritical. However, the technological problems associated with a sodium-heated supercritical steam generator appeared large enough to plan for no more than 2400 psi for the early 1980's. Unfortunately, the present study is somewhat biased by the rather arbitrary limitation that the plant be designed for 1000 MWe output. There is ample evidence to suggest that much larger units will be needed to compete in the 1980 period (indeed, water reactors are now being offered at 1500 MWe net output). The incentive for higher steam pressure at larger plant output would be greater. For the present study, despite possible modest incentives for higher pressure, an upper limit of 2400 psia was used in evaluating different designs, and this was used as the Task II reference steam pressure. This is not meant to imply that development effort should not be extended to study supercritical pressures.

Another example of a technological limit which was not completely explored in the current work is the coolant outlet temperature from the core. Relationships were derived to describe the variation of fuel cycle costs with coolant outlet temperatures (for steady-state operation), but it was found that no convenient basis could be established for showing detailed capital cost variations with increasing outlet temperature conditions. It is expected that fuel cladding will be more limiting than most of the core structural components because of the way the components were designed to minimize thermal stress problems. Nevertheless, a better understanding of thermal gradients and transient conditions in the structure could cause a slightly lower optimum outlet temperature to be selected than the 1150°F mixed-mean shown for the Task II reference.

Of the various limitations discussed above, the technological limitation has the largest effect on the accuracy of the fuel cycle costs. For example, technological uncertainties could delay the growth of the fast reactor after its introduction. This in turn would affect the size of the fuel business, which is one of the most significant factors in establishing fuel fabricating and reprocessing costs. These fabricating and reprocessing costs are also affected by other technological uncertainties. To account for all of these kinds of uncertainties, the Task III effort examined costs of numerous design alternatives (which represented other technological limitations) for a "conservative" cost basis and an "optimistic" cost basis. The optimistic cost basis assumed very advanced fabrication and reprocessing facilities with a large throughput (equivalent to 10,000 MWe of fuel or more per year) and a 10% charge for money. The conservative costs arbitrarily doubled the fabrication and reprocessing unit costs and increased the charge rate to 13%. All costs were computed as a function of plutonium value, with optimization effected at a plutonium value of \$10/gm.

For the Task II reference, this range of costs is shown for certain expected fuel performance characteristics, and in addition a breakdown of costs is provided for a "reference" point between the conservative and optimistic criteria (at \$10/gm fissile Pu).

All three of the limitations discussed previously affect the accuracy of the capital cost estimate. To minimize uncertainties and difficulties in interpreting the capital cost estimate, the following five steps were taken:

1. Relatively small cost groupings were used for comparison with other nuclear plants,
2. High, low and intermediate unit cost values were assigned for estimating key components in the nuclear steam supply system,
3. Where incomplete piping and auxiliary system details proved troublesome, comparison was made with the Demonstration Plant design and cost estimate,
4. Comparison was made by Bechtel and General Electric of similar size water reactor plant costs, and
5. A contingency was incorporated which is commensurate with the limited level of detailed design and with the uncertain technological state.

In summary, many of the limitations of the Task I work have been reduced or eliminated from the work reported herein. A technically feasible and economically attractive plant has been conceptually designed. However, numerous limitations still exist. Where possible, the cost data has been generated to minimize the effect of these limitations. Further, more detailed study and/or development work could demonstrate the need for design

parameters different from the parameters selected for the Task II reference concept.

1.4 CONTRACTUAL OBLIGATIONS (A. S. Gibson)

This document represents partial fulfillment of certain contractual obligations for the 1000 MWe Follow-On Work. In particular, it completes the reporting requirement for the Task II effort. The major objectives of this work were:

1. To prepare conceptual system design descriptions for the final reference concept, and
2. To determine costs of the reference concept.

The system design descriptions were prepared in a level of detail commensurate with the importance of the system (from a technological or economic point-of-view), and commensurate with the state-of-technology of the system being described. Where definition is lacking because of technological limitations, development programs will be suggested in the Task IV report to improve the design definition and to reduce technological uncertainties.

Concerning the system design descriptions, a guide for preparing these descriptions was provided at the inception of the contract. This guide was contained in Appendices II, IV, VII and Ia of Appendix C, "Description of Design and Analytical Services to be Performed for the 1000 MWe Fast Breeder Reactor Follow-On Study Program," dated April 18, 1966. Appendix Ia included a sample outline of final system descriptions to be provided for the detailed design of the systems. Included in this outline was a format with the following basic headings:

1. Introduction
 - 1.1 System Functions
 - 1.2 Summary Descriptions
 - 1.3 System Design Requirements
2. Detailed Description of System
 - 2.1 Components
 - 2.2 Instruments, Controls, Alarms and Protective Devices
3. Principles of Operation
 - 3.1 Startup
 - 3.2 Normal Operation
 - 3.3 Shutdown
 - 3.4 Special or Infrequent Operation
 - 3.5 Emergency
4. Safety Precautions
 - 4.1 Hazards
 - 4.2 Precautions
5. Maintenance
 - 5.1 Preventative Maintenance
 - 5.2 Troubleshooting

Appendices

Schematic Diagram

For the most part, this outline was followed in the preparation of the system descriptions in this report. However, the level of information provided was minimum in some cases, for two reasons:

1. The conceptual design nature of the work did not lend itself to this arrangement or level of detail, and
2. The preparation of Section 1.3, System Design Requirements implies somewhat arbitrary parametric selections or previous optimization.

The first reason noted above resulted from the limited nature of the contract. Since a primary objective of the contract was to focus on development needs and economic incentives, by necessity, many of the system descriptions which would be defined in considerable detail for a final plant design were defined only in sufficient detail to meet the primary objectives of the contract. To obviate problems in this area, fairly large groupings were utilized to define "systems." Even with these large groupings, in some cases it was not appropriate to have an item by item match-up with the suggested format.

Regarding the "system design requirements" suggested as part of the format for the system descriptions, a basic difficulty is involved in defining what a system design requirement is. In a broad sense the only proper system design requirements are, in effect, the functions and objectives for which the system is intended. In this sense, the most appropriate design requirements were specified as the "general system operational requirements" of Appendix 1. For example, these general requirements specified that the plant should be designed as a safe, reliable, and economically competitive, investor-owned, nuclear power central station, upon assumed 1980 technology.

Beyond the statement of design requirements similar to those indicated above, design features and parameter values are chosen to satisfy these design objectives in an optimum manner. With sufficient time to perform optimization studies and with sufficient definition of the overall design objectives there is no need to state any further design requirements. For example, the choice of a particular type of turbine, a particular fuel pin diameter, a particular type of refueling method or a particular level of activation of the secondary sodium are all based on the expectation that these are the economically optimum choices. These are really not design requirements.

Similarly, in the matter of safety, particular design measures should be taken only if these are felt to be the optimum means of achieving a plant which has an acceptable level of defined risk. Given sufficient time and knowledge there are then no real design requirements, only a few broad design objectives.

The above considerations show that the vast majority of items in any list of system design requirements are, in fact, choices based either upon optimization studies or upon a judgment that the proposed value or design alternative will be found to be optimum.

The design requirements utilized in the Task II report are quite general in nature for the reasons outlined above. However, where requirements were needed to limit the work to manageable proportions they were included on the basis of judgment and experience. Typical of such arbitrary requirements are the following:

1. Use of mixed oxide fuel,
2. Use of stainless steel structural material, and
3. 5200°F maximum fuel temperature.

Particular ground rules were suggested or made mandatory as part of the contract with regard to the method of performing certain calculations which ultimately yielded cost information. These ground rules will be discussed in some detail in Section 1.6. In general, definition of many of the technical data which strongly affect economics was made the responsibility of the contractor. Where these data are different from suggested data provided by Argonne National Laboratory, the contractor has the responsibility of justifying his selection, and in some cases, of performing additional computations for comparative purposes using the Argonne suggested data. Two instances where the General Electric data are different from the Argonne data or ground rules are the nu-

clear cross section data and the input feed plutonium isotopic composition data. The suggested Argonne cross section data utilizes the "MC²" system which is a derivative of the ENDF/B cross section library. As discussed earlier, recent experiments provided evidence that the plutonium alpha values should be higher than those used in ENDF/B. Additionally, the specified plutonium feed isotopic data provided by Argonne is more closely related to fast reactor discharge plutonium than to water reactor discharge plutonium. Based upon business studies conducted in General Electric and elsewhere, water reactor plutonium appears to more closely represent what might be available in the 1980-1990 period. Therefore, higher plutonium alpha values and water reactor plutonium feed were used in the Task II computations. To satisfy the requirements of the contract, a comparison of the important reactor characteristics and economics will be made in the Task III report with the Argonne suggested data.

In summary, the basic philosophy followed in meeting the contractual obligations for the Task II effort, was to provide General Electric's best estimate of how a commercial LMFBR might be conceptually described and what its costs might be in the 1980 period if the R&D described in the Task IV report is accomplished. This description and costs are based on extrapolation of the present technology and knowledge and could change as the technology develops. In some cases, strict adherence to the philosophy of showing what General Electric believes the situation might be in the 1980 period was not possible because of explicit restrictions in the contract. A case in point is the size limitation of 1000 MWe; larger plants are expected to be commercially desirable in that period. In instances where the contract did not expressly forbid application of the basic philosophy, it was followed, and in some cases additional data will be provided in the Task III report to show comparative results with the contract specified guidance data.

1.5 SUMMARY PLANT DESCRIPTION

(A.S. Gibson)

The reference power plant is an indirect-cycle, sodium-cooled fast reactor with a tandem-compound 6-flow reheat turbine. Net power output from the power plant is 1011 MWe. Figure 1.1 shows a cutaway view of the reactor and refueling cell. As can be seen from this figure, the reactor primary pumps, intermediate heat exchangers and interconnecting piping are all located in a single sodium-filled primary vessel. This primary vessel is of stainless steel construction and is 48 feet in diameter. Surrounding the primary vessel is a secondary carbon steel vessel which is insulated. Outside of both vessels is the concrete shielding with a steel liner. These various vessels and liner are utilized to maintain coolant integrity in the event of a vessel failure.

Three primary sodium pumps and three intermediate heat exchangers are mounted on the upper shield deck, and the primary vessel is supported from this same concrete shield deck. As seen from Figure 1.1, the reactor core and vessel are also supported from the shield deck, and are located off-center with regard to the primary vessel.

One of the intermediate heat exchangers is shown in the raised position in Figure 1.1. Each of the pumps and heat exchangers is designed to be removed into a cask for removal to a maintenance cell. The cask is not shown in the figure to simplify the illustration.

Refueling of the reactor is accomplished by raising the shield plug inside of a hot cell which has an inert atmosphere of argon. Figure 1.1 shows the shield plug in the raised position. Automated fuel shuffling and transfer mechanisms provide for transport of the new fuel from a storage tank in the hot cell into the reactor, and for removal of used fuel into a second storage pit for decay and later shipment. With the shield plug in place

over the reactor, and a separate shield plug over the fuel decay tank, personnel can enter the hot cell in specially designed suits for maintenance of equipment. Control rods and instrument probes are removable from the top of the shield plug for maintenance. Fuel transfer and shuffling mechanisms can also be repaired or maintained by contact maintenance in the hot cell, or components can be removed and replaced.

During normal operation of the reactor, a controlled cover gas pressure of argon is maintained over the sodium in three separate regions of the primary system vessel. A buffer seal arrangement at each penetration of the shield deck or shield plug prevents leakage of the cover gas into the reactor building or the hot cell. Cooling in the secondary vessel and shield concrete is provided by a circulating nitrogen system.

Figure 1.2 shows a cross section through the entire power plant. The reactor, refueling cell, and reactor support systems (sodium storage, cold traps, gas handling systems) are located in a reinforced concrete structure. A concrete wall rises approximately 28 feet above grade from this structure to protect primary system components from tornado generated missiles. Resting on and attached to this concrete wall is a structural steel and metal panel superstructure with a tar and gravel roof. The total concrete-structural steel complex constitutes the reactor building. It is approximately 105 feet wide by 204 feet long, and extends 105 feet above grade and 50 feet below grade. Located within the reinforced concrete structure are individual steel lined cells for housing critical nuclear components such as the reactor, cold traps, and the radioactive gas system. For sodium systems, the cells are filled with nitrogen to minimize the possibility of a sodium fire. The cells are designed for low leakage. All of these cells and nuclear equipment are located within the inner rooms of the concrete structure

- 1 REACTOR CORE
- 2 REFUELING CELL WALL
- 3 SHIELD PLUG (IN RAISED POSITION)
- 4 CONTROL ROD DRIVES
- 5 SHIELD PLUG LIFTING SCREW & GUIDE
- 6 FUEL TRANSFER MACHINE
- 7 FUEL SHUFFLING MACHINE
- 8 FUEL DECAY TANK
- 9 NEW FUEL STORAGE
- 10 INTERMEDIATE HEAT EXCHANGER (3 UNITS)
- 11 INTERMEDIATE HEAT EXCHANGER (SHOWN IN RAISED POSITION)
- 12 SECONDARY SODIUM INLET
- 13 SECONDARY SODIUM OUTLET
- 14 PRIMARY SODIUM PUMP AND DRIVE MOTOR (3 UNITS)
- 15 PRIMARY SODIUM VESSEL
- 16 INSULATION AND OUTER TANK
- 17 STEEL LINER ON CONCRETE
- 18 CONCRETE SHIELDING

FIGURE 1.1 CUTAWAY VIEW OF THE REACTOR AND THE REFUELING CELL

FIGURE 1.2 PLANT ELEVATION-SECTION B-B
(SIMILAR TO BECHTEL DRAWING M-6)

so that there is no danger of ground water seeping through an outside concrete wall to the steel liner. The superstructure and above grade portion of the reactor building (and below grade in peripheral rooms) has a controlled heating and ventilation system. In the event of leakage of radioactivity into the area, the building effluent is discharged through a filtering-scrubbing system and up the stack.

Adjacent to the reactor building are the steam generating building structure, the turbine auxiliary building and the turbine building. The turbine building is the largest structure in the plant; it is 300 feet long by 108 feet wide by 104 feet tall. Pipeways are provided where necessary within all buildings and are designed to facilitate maintenance of equipment and to yield adequate room for thermal expansion loops of the piping. Trenches are provided within the steam generator building and are filled with METL-X or other fire retardants to minimize the possibility of a disabling secondary sodium system fire in the event of a sodium leak.

The steam generators are designed for removal to a centrally located shop for maintenance. In the event of a water leak into the sodium side of the steam generators, rupture discs are provided to vent the reaction products. Quick action valves isolate the water-steam systems from the steam generator units to minimize the total sodium-water reaction.

The turbine generator is a tandem-compound, 6-flow reheat machine with steam conditions of 2400 psia, 950°F/950°F. It is arranged in the turbine building so that the 150-ton service crane can service all major components of the turbine generator, and additionally can service other critical turbine support equipment such as the high pressure feedwater heaters.

Figure 1.3 shows the plant heat balance and schematic flow diagram. It should be noted that although this diagram shows one coolant loop, this is schematic only. As previously observed, three separate primary system pumps and intermediate heat exchangers are provided. There are also three separate secondary sodium loops with independent pumps and steam generators for each loop. The steam generators include two boiler units, one superheater unit and one reheater unit for each of the three loops. Similarly, there are three separate feedwater drains for the turbine. Thus Figure 1.3 shows the total flows and heat transferred for all of the loops and feedwater drains in the plant. A net plant efficiency of 41.7% is shown in the figure. This relatively high efficiency is made possible by the 1150°F sodium outlet temperature from the core, and by the sodium-to-steam reheater. Because of the excellent heat transfer characteristics of sodium, even though a double sodium fluid heat transport arrangement is used between the reactor and the steam generators, steam temperatures of 950°F are obtained from both the superheater and the reheater.

Table 1.1 shows some of the major plant characteristics and parameters for the reference concept. As noted from the table, the reactor core is "non-spoiled" (no BeO addition, no extra pancaking or modularization) and utilizes mixed-oxide fuel. The fuel pins are of the vented type. Steel volume fraction in the core is kept low (<15%) by using vented fuel pins and non-free-standing channel walls. Because of the low steel fraction (and high fuel volume fraction), and because of the optimum geometry, the core uses neutrons relatively efficiently and has a high internal conversion ratio and breeding ratio. The inventory requirements are also low for the same reasons. By operating the 1/4-inch fuel pins with a central void (achieved during initial operation by sintering) and at

a temperature in the peak pin near melting; and by maintaining a peak cladding temperature of 1300°F, a high specific power is attained (157 kW/kg U + Pu and 1304 kW/kg fissile). All of these effects result in the relatively low doubling time shown of 7.75 years. Moreover, the high internal conversion ratio makes possible the attainment of an annual refueling schedule at 80% capacity factor with modest control requirements (12 burnup or shim control rods; 31 total rods in core). The fuel is designed to achieve an average burnup of approximately 100,000 MWd/Te U + Pu.

From Table 1.1 and from Figure 1.1 it can be seen that blast shields to absorb large quantities of energy during a hypothetical fuel melt-down or large sodium voiding accident are not provided within the primary vessel structure. The blast shields are unnecessary because very large accidents of the type described are prevented from occurring through the use of core safety instrumentation and protective systems. On Figure 1.1 many of the instrument probes, which are positioned over each fuel channel, can be seen on the under-side of the raised reactor shield plug. It is anticipated that each of these probes could contain multiple thermocouples for coolant temperature measurement; other flow measuring probes could be located at the bottom of each fuel channel near the lower grid structure shown in Figure 1.1. Because of the early state of the present technology, it is not possible to explicitly define the measuring devices or probes which will be used. A number of different devices show promise of being successfully developed. Illustrative examples are included in this report, and development programs will be described in the Task IV report. Because of the numerous devices possible which have promising characteristics, the probability for development of sufficient devices to make the basic safety philosophy a successful one is very high.

Automatic control of load change on the reference concept is effected over a range from 20% of full load to full load. To mini-

mize thermal shock of the fuel cladding and reactor structural components during load changes, the reactor coolant outlet and inlet temperatures are maintained approximately constant as the load is varied. This is accomplished by varying the primary coolant flow rate (by pump speed control). Flow rate of the secondary sodium coolant is also varied with load. Since the sodium inlet and outlet temperatures from the reactor are essentially constant, the steam temperatures from the superheater and reheater vary as the load changes. Before steam is admitted to the turbine, its temperature is attemperated as needed by adding desuperheating sprays or by bypassing lower temperature steam around the reheater. No large sodium control valves are needed for this method of control. The control logic and process monitoring is accomplished in three major computers: A digital plant control computer; a digital reactor control computer; and an analog reactor control back-up computer. The plant control computer measures plant demand from the area dispatch controller and adjusts all system controllers accordingly. These controllers in turn set valve positions, control rod positions, and pump speed as needed. Except for safety system trips, shutdown of the reactor is not by scram. In order to minimize thermal shocks caused by rapid power and temperature swings, shutdown is accomplished by gradually reducing power and temperature.

Following normal shutdown of the reactor, removal of decay heat from the core is accomplished by using the normal plant heat transfer system at reduced power. Any one of the three sodium coolant loops can provide for decay heat removal. The pumps are operated at reduced speed by means of pony motors. Steam generated in the evaporators is discharged to the main condenser where final heat rejection is carried out. Only one of the two main circulating water pumps is used for this purpose.

GENERATOR
338,352 KW
90 PF
1.0 PSIG H₂ PRESSURE
753 KW GEN. LOSSES
167 KW FIXED LOSSES

14

TABLE 1.1
TASK II REFERENCE* REACTOR:
TENTATIVE PLANT CHARACTERISTICS

A. GENERAL

1. Geometry, inches

Core height	30.0
Axial blanket thickness (each end)	15.0
Radial blanket thickness	10.5
Core diameter (equivalent cylinder)	97.4

2. Fuel Type

Cladding material	316 Stainless-steel
Fuel material	(U, Pu)O ₂
Gap material	Gas
Plenum type	Vented

3. Fuel Subassembly Arrangement

No. of pins/subassembly (without control rod)	
Fuel	331
BeO	0
No. of subassemblies in core	
Without control rods	234
With control rods	31
Subassembly channel wall type	Non-freestanding
Subassembly hold down	Hydraulic (mech. stop)

4. Control Rods

Burnup control	
Material	Tantalum
Number	12

* The choice of these particular options was made only to allow the definition of a specific reactor plant design for the purposes of developing a meaningful cost estimate and to provide a consistent framework in which to evaluate the alternative options. The incentive for incorporating the alternatives is discussed in the Task III report of this design study.

TABLE 1.1 (Continued)

Scram control		
Material	Enriched B ₄ C	
Number	(20-25% B-10)	
	19	
5.	<u>Vessel and Primary System Arrangement</u>	
Primary system location	Primary sodium tank	
Primary sodium tank		
Material	Stainless steel	
Diameter, feet	48.0	
Overall height, feet	40.0	
Containment vessel		
Material	Carbon Steel	
Diameter, feet	51.3	
Reactor vessel		
Material	Stainless steel	
Diameter, feet	15.3	
Blast shields	Not required	
6.	<u>Refueling Arrangement</u>	
	Hot cell	
7.	<u>Fuel Mass, kg</u>	
	<u>Oxide</u>	<u>Heavy Metal</u>
Core	15,261	13,452
Axial blanket	17,057	15,035
Radial blanket	18,926	16,683
Total	51,244	45,170
8.	<u>Sodium Inventory</u>	
Primary heat transfer system	3.0×10^6 , lbs	
Secondary heat transfer system	0.7×10^6 , lbs	
9.	<u>Inert Gas Systems</u>	
Reactor cover gas	Argon	
Containment vessel and cold trap vaults	Nitrogen	

TABLE 1.1 (Continued)

B. CORE DATA AND PERFORMANCE

1.	<u>Power Output (mid-cycle), MWt</u>	
	Core	2118
	Axial blankets	183
	Radial blanket	116
	Total	2417
2.	<u>Specific Power (core average), kW/kg</u>	
	U + Pu	157
	Fissile (mid-cycle)	1304
3.	<u>Nuclear Performance (mid-cycle)</u>	
	Fissile enrichment (zone 1/zone 2), %	10.73/13.45
	Core fissile load, kg Pu 239, 241	1624
	Blanket fissile load, kg Pu 239, 241, 235	611
	Breeding ratio	1.42
	Core conversion ratio (zone 1/zone 2)	1.01/0.76
	Doubling time, yr	7.75
	Fast neutron flux, n/cm ² -sec (>0.825 MeV)	
	Peak	1.5×10^{15}
	Average	0.976×10^{15}
	Cladding fluence (at the design limiting location) >1MeV	$\sim 6.0 \times 10^{22}$
	Peak fuel cladding total fluence >0.825 MeV	7.42×10^{22}
	Peak fuel cladding total fluence	56.4×10^{22}
	Doppler coefficient (-T dk/dT)	
	Sodium in	0.0060
	Sodium out	0.0037
	Sodium void reactivity effect, (1)\$	
	Total core and blankets	2.24
	"Maximum"(2)	5.20
	One bundle "maximum"(3)	0.06
	Reactivity control requirements, \$	22.26
	Refueling schedule (at 80% load factor), months	12
	Average core fuel in-pile time, days	790
4.	<u>Core Composition (mid-cycle), volume %</u>	
	Fuel	46.7
	Sodium	38.0
	Steel	14.9
	BeO	---
	Tantalum	0.4

(1) $\beta_{\text{eff}} = 0.0038$

(2) Based upon partial voiding outward from core center to 2/3 of core height and 80% of core volume in radial direction

(3) Based upon one subassembly at the center of the reactor

TABLE 1.1 (Continued)

5. Fuel Pin Geometry and Data (Cold Dimensions)

Diameter (o.d.), inches	0.248
Cladding thickness, inches	0.0105
Pin array	Triangular
Pin pitch/diameter ratio	1.20
Helical wire wrap spacer pitch length, inches	6
Fuel density (hot smeared), % theoretical (10.5gm/cc)	85
Average core discharge exposure, kWd/kg(U + Pu)	100,167

6. Thermal-Hydraulic Performance (mid-cycle)

Peak data (at 1.10 times rated power)	
Max. fuel temperature, °F	5200
Linear power generation, kW/ft	17.9
Surface heat flux, MBtu/h-ft ²	0.93
Average data (at rated power)	
Linear power generation, kW/ft	9.63
Surface heat flux, MBtu/h-ft ²	0.502
Subassembly pressure drop, psi	64
Peak coolant velocity, ft/sec	25.4
Peak coolant outlet temperature at full power, °F	1290
Peak cladding temperature at full power, °F	1300
Peak design point cladding temperature at full power, °F	1225
Mixed mean coolant outlet temperature, °F	1150
Core coolant temperature rise, °F	340

7. Core Peaking Factors

Power	
Axial distribution	1.20
Radial distribution	1.22
Control rod flux depression	1.05
B.O.L. to M.O.L. shift	1.05
Hot spot (uncertainties)	1.05
Overpower allowance	1.10
Total product	1.86
Coolant Enthalpy	
Radial power distribution	1.22
Control rod flux depression	1.05
Subassembly B.O.L. to M.O.L. shift	0.98
Subassembly flow distribution effect	1.12
Effect of orificing (see Section 2.2.3.2)	0.90
Hot stream (uncertainties)	1.06
Coolant dilution (leakage and radial flow)	1.05
Total product	1.41

TABLE 1.1 (Continued)

C. PLANT DATA AND PERFORMANCE

1. Plant Performance

Gross power output, MWe	1038
Net power output, MWe	1008
Net heat rate, Btu/kW-h	8192
Net plant efficiency, %	41.7

2. Systems Arrangement

Primary heat transfer system	
Number of primary pumps	3
Number of intermediate heat exchangers	3
Primary fluid location in IHX	Tubes
Secondary heat transfer system	
Number of loops	3
Number of pumps	3
Steam Generators	
Type	Once-thru
Cycle	Reheat
Arrangement	Parallel reheat and superheat
Number of Units	
Superheater	3
Reheater	3
Evaporator	6
Turbine-Generator	
Type	Tandem-compound, 6-flow reheat
Rpm	3600

3. Systems Data

Primary heat transfer system	
Total mass flow rate, 10^6 lb/h	80.6
Reactor outlet temperature, °F	1150
Reactor inlet temperature, °F	810
Total pressure drop, psi	87
Secondary heat transfer system	
Total mass flow rate, 10^6 lb/h	64.3
IHX outlet temperature, °F	1071
IHX inlet temperature, °F	646
Pressure drop, psi	92
Main steam conditions	
Total mass flow rate, 10^6 lb/h	7.265
Turbine throttle temperature, °F	950
Turbine throttle pressure, psia	2400

TABLE 1.1 (Continued)

Reheat steam conditions	
Total mass flow rate, 10^6 lb/h	6.560
Reheat outlet temperature, °F	950
Intermediate turbine inlet pressure, psia	540
Feedwater temperature, °F	481
Turbine exhaust pressure, inches of Hg abs.	1.5
Number of feedwater heaters	7

D. REACTOR BUILDING

1. Arrangement and Type

Shape	Rectangular
Dimensions, feet, length/width	204/105
Height (total)	105
Design pressure, psig	0
Materials	Insulated siding and structural steel

2. Refueling Cell

Shape	Rectangular
Materials	Reinforced concrete
Design pressure, psig	0

In an emergency, station electrical power may be lost. During such an emergency a separate decay heat removal system is used which requires less electrical power than the normal shutdown cooling system. Instead of rejecting heat to the main condenser, steam from the evaporator is discharged to a high pressure condenser which rejects its heat to the plant service water system. Small pumps provide the pumping power for the evaporator and the service water system. These water pumps and the pony motors for the sodium pumps are connected to the emergency electrical power supply. The emergency power is supplied from two major bus systems. One system is a "short-break" system for equipment which can tolerate a short interruption of

power. This system derives its power source from two of three half-capacity diesel generators which can start automatically in 20 seconds. The second emergency power system is a "zero-break" system for equipment which cannot tolerate a short interruption of power (e.g. control rod operators). Power for this system is provided through the station batteries. Battery power is used to drive two of three half-capacity motor-generator sets which generate ac power. When the station power is available, the motor-generator sets are kept in a spinning condition by a small ac motor. A flywheel keeps the units spinning upon loss of station power until the dc motors pick up the load.

Table 1.2 lists the major plant equipment.

TABLE 1.2

PLANT MAJOR EQUIPMENT LIST

<u>Item</u>	<u>Number</u>
Reactor:	
Fuel Bundle, Inner Zone (without control rod)	108
Fuel Bundle, Outer Zone (without control rod)	126
Fuel Bundle, Inner Zone (with B_4C control rod)	19
Fuel Bundle, Outer Zone (with Tantalum control rod)	12
Radial Blanket Bundle	126
Reflector Guide Bundle	72
Core Support Structure	1
Shim Rod Drive (includes 6 startup rod drives)	18
Scram Rod Drive	13
Refueling Cell:	
New Fuel Storage Tank	1
Irradiated Fuel Storage Tank	1
Fuel Shuffling Machine	1
Fuel Transfer Thimble	161
Fuel Transfer Grapple	1
Bridge Crane Mounted Telescoping Tube Manipulator	2
Auxiliary Service Mechanism	1
Shield Plug Hoist	1
Primary Heat Transfer System:	
Primary Sodium Tank	1
Primary Containment Vessel	1
Reactor Vessel	1
Intermediate Heat Exchanger (IHX)	3
Primary Sodium Pump and Drive	3
IHX Transfer Container	1
Pump Transfer Container	1
Secondary Heat Transfer System:	
Secondary Sodium Pump and Drive	3
Water Evaporator	6
Steam Superheater	3
Steam Reheater	3
Expansion Tank	3

TABLE 1.2 (Continued)

Condensate System:

Main Condenser	1
Air Ejector	3
Air Ejector Desuperheater	1
Condensate Pump and Drive	2
Condensate Storage Tank	1
Condensate Demineralizer	6
Condensate Transfer Pump and Drive	1
Gland Seal Condenser	1

Feedwater System:

Boiler Feed Pump and Turbine Driver	2
Boiler Feed Pump and Drive Lube Oil Conditioning System	2
Deaerator	1
Low Pressure Feedwater Heater	9
High Pressure Feedwater Heater	9

Steam System:

Steam Turbine (TC-6F, 33-1/2" LSB)	1
Steam Turbine Lube Oil Conditioning System	1
High Pressure Turbine Stop Valve	2
Intermediate Pressure Turbine Stop Valve	2
High Pressure Turbine Throttle Valve	4
Generator (22,000 V ac, 1100 MVA, 3600 rpm)	1
Hydrogen System	1
Exciter (Silicon Diode Rectifier)	1

Water Storage and Makeup System:

Clarifier Tank	1
Chemical Feed System	1
Clearwell Tank	1
Transfer Pump and Drive	3
Filter	3
Cation Exchanger	2
Anion Exchanger	2
Mixed Bed Ion Exchanger	2
Acid and Caustic Regeneration System	1

Service and Circulating Water System:

Circulating Water Pump and Drive	2
Traveling Screen and Drive	4

TABLE 1.2 (Continued)

Screen Wash Pump and Drive	2
Chlorine Injection System	1
Service Water Pump and Drive	3
Auxiliary Cooling System:	
Decay Heat Condensate Pump and Drive	2
Decay Heat Condenser	1
Sodium Service System:	
Radioactive Sodium Cold Trap	4
Non-Radioactive Sodium Cold Trap	6
Economizer	10
Expansion Tank	2
Pump (Electromagnetic)	2
Plugging Indicator	2
Filter	1
Radioactive Sodium Storage Tank	1
Non-Radioactive Sodium Storage Tank	1
Steam Generator Pressure Relief System:	
Boiler Inlet Stop Valve (Quick Acting)	3
Superheater Outlet Isolation Valve	3
Boiler Relief Valve	3
Reheater Relief Valve	3
Rupture Disc	24
Irradiated Fuel Storage Cooling System:	
Na-to-NaK Coiled Tube Heat Exchanger	2
Pump (Electromagnetic)	2
NaK-to-Air Heat Exchanger	2
NaK Expansion Tank	1
NaK Transfer Tank	1
NaK Drain Tank	1
NaK Plugging Indicator	1
NaK Cold Trap	1
Nitrogen System:	
Nitrogen Supply Tank (Cryogenic)	2
Nitrogen Cooler (Freon)	3
Nitrogen Dryer (Freon)	2
Nitrogen Cooler (Water)	2

TABLE 1.2 (Continued)

Nitrogen Blower (90 hp)	3
Nitrogen Blower (3 hp)	2
Nitrogen Filter	3
Argon-To-Nitrogen Heat Exchanger	4
Argon System:	
Argon Supply Tank (Cryogenic)	2
Refueling Cell Pressure Control Tank	1
NaK Bubbler	2
Argon Cooler (Freon)	2
Argon Compressor (Refueling Cell Gas)	4
Argon Compressor (Reactor Vessel Gas)	3
Argon Compressor (Primary Sodium Tank)	2
Argon Compressor (IHX Gas)	4
Radioactive Gas Handling System:	
Argon Blower (Primary Sodium Tank Exhaust)	2
Nitrogen Blower (Equipment Cell Purging)	2
Radioactive Gas Compressor (Storage Tanks)	4
Air Blower (Exhaust Stack)	2
Radioactive Gas Cooler	1
Radioactive Gas Storage Tank	4
Electrical Systems:	
Generator Control Panel	1
Metering Equipment	-
Generator-Isolated Phase Bus	1
Auxiliary Power Transformer Isolated Phase Bus	1
Auxiliary Power Transformer (22 kV/13.8 kV)	1
Startup and Standby Transformer	1
Auxiliary Power Transformer (13.8 kV/480 kV)	3
Auxiliary Power Transformer (480 V/120 V-280 V)	2
Switchgear (13.8 kV)	2
Switchgear (480 V)	2
Switchgear (120 V)	2
Switchgear (125 V dc)	2
Switchgear (250 V dc)	1
Battery Charger	3
Battery (900 Ampere-Hour)	2
Motor-Generator Set (120 V-280 V)	3
Motor Control Center (480 V)	12
Diesel Generator (1500 kW)	3
Fuel Storage Tank	1
Diesel Fuel Transfer Pump and Drive	2

TABLE 1.2 (Continued)

Heating, Ventilating and Air Conditioning System:

Reactor Building Fan and Drive	11
Steam Generator Building Fan and Drive	2
Turbine Building Fan and Drive	12
Emergency Generator Building Fan and Drive	1
Administration Offices Fan and Drive	2
Control Room Fan and Drive	1
Warehouse and Shop Fan and Drive	2
Absolute Filter System	2
Sodium Oxide Scrubber	1
Air Conditioning Chiller	1

Plant and Instrument Air System:

Air Compressor, Drive, and Aftercooler	3
Air Receiver Tank	3

Fire and Domestic Water Systems:

Water Pump and Drive (Electric)	1
Water Pump and Drive (Diesel)	1
Jockey Pump and Drive	1

Breathing and Cooling Air System:

Air Compressor, Drive, and Receiver	2
Air Exhaust Vacuum Pump and Drive	2
Air Refrigeration System	4
Air Receiving Tank	2
Breathing Air Hose Connection	336

Plant Cranes and Hoists:

Turbine Building Bridge Crane	1
Reactor Building Bridge Crane	1
Machine Shop Monorail System and Hoist	1
Warehouse Hoist	1

Instrument and Control System:

Core Thermocouple Probes (4 T/C each)	256
Core Flow Measuring Venturis and Level Probes	256
Miscellaneous Safety and Control Instruments	-
Reactor and Plant Control Computers	3
Neutron Monitoring System For Reactor Control	1
Plant Radiation Monitoring System	1

1.6 ECONOMICS (A. S. Gibson)

Table 1.3 shows the total energy costs for an early commercial 1000 MWe LMFBR. Comparable costs were obtained from Public Utility Commission records of a recent pressurized water reactor. These records imply a capital cost of 158 \$/kWe and a total energy cost of 4.94 mills/kWh. Some adjustment to the capital costs of the water reactor in the PUC records was made to allow for escalation to 1968 dollars. The fuel cycle costs for this water reactor is 1.67 mills/kWh. This is not based upon the same economic ground rules as for the LMFBR. The LMFBR utilizes a higher annual charge rate for non-depreciating capital and a higher plutonium value than does the water reactor. If applied to the water reactor, the higher annual charge rate would result in a higher fuel cost than that shown and the higher fissile plutonium value would have the effect of lowering these costs. It was beyond the scope of the present contract to analyze comparative water reactor costs. They are presented merely as a point of uncertain reference.

The "reference" costs shown for the early commercial LMFBR are based upon the "expected" capital costs discussed in Section 1.6.1 and the "reference" fuel cycle costs discussed in Section 1.6.3. The operating and maintenance costs shown are those generated in Section 1.6.2. Some of the costs are quite dependent upon the economic assumptions specified in the contract and upon specific economics models used. The fuel cycle costs, for example, utilize a fairly optimistic assumption concerning the throughput of the fabrication facility and the state of technology during this period. The "reference" fuel cycle costs, for example, assume a very high fabrication and reprocessing plant throughput which is equivalent to reload 10,000 MWe of installed reactor capacity and an advanced fabrication technology. These costs of 0.38 mills/kWh would therefore be associated with a strong and mature industry possibly corresponding to

the majority of the 35 year life of the reactor. During the early life of the reactor, considerably higher costs could be expected, even under optimistic conditions. These higher costs could approach the "high" values shown of 0.75 mills/kWh.

To obtain a feel for alternative assumptions, an attempt was made to define upper and lower bounds on the cost of an early commercial 1000 MWe LMFBR. These are designated in Table 1.3 as the high and low costs respectively. The high costs utilize the "90% probability" values shown for capital costs and discussed in Section 1.6.1 and the fuel costs utilize the "conservative" economic basis of Section 1.6.3. Similarly, for the low costs the capital costs are based upon the "10% probability" values of Section 1.6.1 and the fuel cycle costs are based upon the "optimistic" economic basis of Section 1.6.3. The operating and maintenance costs in the high and low columns merely reflect an estimate of upper and lower limits based on different utility operating practices.

1.6.1 Capital Costs (A. S. Gibson, W. F. Garetson)

Capital costs are presented in the following material for the reference plant described in this Task II report. No attempt is made in this report to describe details of the methods used to determine capital costs for each of the component items in the plant. In general, previous experience was relied upon to determine unit costs which could be applied to the separate components. For example, the costs of the reactor vessel were determined by multiplying the weight of the unit by a specific cost per unit weight determined from previous experience on analogous structural components. For components such as the steam generators a series of cost coefficients were derived to represent the different labor and material items making up the steam generators. These coefficients were applied as multipliers to particular design parameters,

TABLE 1.3
TOTAL ENERGY COSTS

	Early Commercial 1000 MWe LMFBR		
	Reference	High	Low
Capital Costs, \$/kWe*	195	229	170
<u>Energy Costs, mills/kW-h**</u>			
Capital	3.63	4.25	3.16
Fuel Cycle	0.38	0.75	0.29
Operating and Maintenance (including insurance)	<u>0.39</u>	<u>0.43</u>	<u>0.32</u>
Total	4.40	5.43	3.77

*Based on 1968 dollars

**At 0.8 capacity factor, 13% capital charge rate for depreciating capital, 10% capital charge rate for non-depreciating capital, and \$10/gm fissile plutonium value.

and the resulting costs were summed. This method of determining costs will be discussed in the Task III report. Building and structural concrete costs were determined from estimates of the unit costs of materials, the quantity of materials, and suitable labor cost multipliers to these material costs (determined by experience on analogous buildings).

In the process of preparing the capital cost estimate it became apparent that the method of approach utilized could play a major role in the outcome of the estimate. Indeed variances in the method of preparing the cost estimate can have a larger effect in producing differences in the capital costs than can the differences produced by selecting alternative reactor types and arrangements.

reactor types and arrangements. The method used is summarized in the following steps:

1. Base all costs on 1968 dollars with no allowance for escalation to 1980,
2. Engineer equipment and systems in sufficient detail to permit costing individual components,
3. Utilize previous experience on fossil-fired and water reactor plants in systems arrangements and design,
4. Provide "three point" cost estimate of key systems,
5. Compare individual cost accounts with published information on water reactors,

6. Obtain "customer costs" from utilities and Bechtel,
7. Base the indirect costs such as field distributables and engineering on previous water reactor experience,
8. Show costs for the "first-of-a-kind 1000 MWe LMFBR" and "an early commercial LMFBR", and
9. Compare the total costs of the LMFBR with the published total costs of a 1000 MWe water reactor.

Table 1.4 shows the summary of the capital costs for three different 1000 MWe reactors. The first column represents the initial 1000 MWe LMFBR to be built for operation in the early 1980 period (but not the first LMFBR, since Demonstration Plants must precede the first 1000 MWe unit). The second column is intended to represent an early commercial LMFBR also operational in the 1980 period. The cost for the two LMFBR's differ only by the amount allowed for contingency and for first-of-a-kind engineering.

For comparative purposes a 1000 MWe water reactor is included. Costs for the water reactor were extracted from California Public Utility Commission records and represent a recent pressurized water reactor with an adjustment to the costs to provide escalation to 1968 (6-1/2%/year on approximately 80% of the total plant cost).

Table 1.5 shows the details of the individual accounts which make up the summary costs shown in Table 1.4. Both Tables 1.4 and 1.5 include reactor manufacturer engineering, G&A and fees distributed within account No. 22.

Table 1.5 also includes three separate columns, respectively entitled "Expected

Value," "10% Probability" and "90% Probability." The 10% and 90% probability values are an attempt to put a high and low limit on the estimated cost. They represent a judgment that the true values will be less than those shown by a probability of 10% or 90% respectively. This type of estimate is sometimes designated a "three point" cost estimate. The individual accounts in the 10% and 90% probability columns were simply added to obtain the total costs. Probability theory would suggest that a new total should be determined based upon the individual account probabilities. This new total would be somewhat less than a simple sum of the individual accounts, or alternatively a higher probability would be assigned to the sum.

Tables 1.4 and 1.5 show a \$12,000,000 allowance for "Incomplete-Design" as a part of the direct construction cost. This allowance is provided for the increase in equipment and system complexity which normally occurs as a plant design progresses from the conceptual stage. This increase was estimated by comparing the conceptual drawings of this plant to the more detailed drawings of similar plants which have progressed beyond the conceptual design phase.

Also shown in Table 1.4 is an item called "Contingency; General Uncertainty." It is larger for the first 1000 MWe LMFBR than for an early commercial LMFBR. It represents an allowance to account for construction uncertainties, delay in schedule, design changes, et cetera.

The customer costs of 20% shown in Tables 1.4 and 1.5 were determined by conversations with Bechtel Corporation and utilities. These costs vary from utility to utility based upon different accounting practices and different site conditions. The minimum cost to be expected might be 18% of the construction costs and a maximum might reach 26%. A representative cost is as follows:

Interest during construction	15%
Utility General and Administrative Expense (G&A)	5%
Ad-Valorem Taxes	2-1/2%
Compound Total	24%

The 20% figure shown in Tables 1.4 and 1.5 was selected as a relatively optimistic but realistic assessment of what these costs might be.

A large cost item shown in Tables 1.4 and 1.5 is account No. 226, "Instrument and Control." Although the specific instrument system shown in the Task II report was used to determine the costs of individual instruments and of the complete system, it should be recognized that considerable uncertainty exists in the definition of this item (see discussion in the Introduction to this report). This is reflected in the "three-point" cost estimate which shows a large variation for this particular account.

TABLE 1.4
1000 MWe REACTOR POWER PLANT CAPITAL COST SUMMARY

(Costs are in 10³ 1968 Dollars)

		First 1000 MWe LMFBR	Early Commercial LMFBR	Water*** Reactor
21	STRUCTURES AND IMPROVEMENTS	14,624	14,624	
22	REACTOR PLANT EQUIPMENT*	76,054	76,054	
221	Reactor Equipment	10,906		
222	Heat Transfer Systems	39,934		
223	Nuclear Fuel Handling	3,399		
225	Radioactive Waste	673		
226	Instrument and Control	13,312		
227	Feedwater Supply and Treatment	3,010		
228	Steam, Cond., and Feedwater Piping	4,820		
23	TURBINE GENERATOR PLANT EQUIP.	24,618	24,618	
24	ACCESSORY ELEC. EQUIPMENT	3,025	3,025	
25	MISC. POWER PLANT EQUIPMENT	1,055	1,055	
	First-of-a-Kind Engineering	15,000		
	Allowance for Incomplete Design	12,000	12,000	
SUB TOTAL; DIRECT CONSTRUCTION COSTS		146,376	131,376	
	Field Distributables	13,441	13,441	
	Engr-Constructor; Engin., Inspec., Fees	10,300	10,300	
	Contingency; General Uncertainty	17,400	8,000	
SUB TOTAL: CONSTRUCTION COSTS		187,517	163,117	139,500
	Customer Costs (Land, Interest During Const., G&A, Ad-Valorem Taxes) 20%	37,503	32,623	27,900
	Sodium Inventory	925	925	
TOTAL PLANT **		225,945	196,665	167,400
	Net Output, MWe	1,010	1,010	1,060
TOTAL PLANT, \$/kWe net		224	195	158

* Includes reactor manufacturer engineering, G&A, fees

** Excludes step-up transformer, transmission, and substation

*** Based on California Public Utility Commission records with adjustments for escalation

TABLE 1.5
CAPITAL COST ESTIMATE FOR
1000 MWe EARLY COMMERCIAL LMFBR
(All Costs in 10³ 1968 Dollars)

		Expected Values	10% Probability	90% Probability
		<u>Total</u>	<u>Total</u>	<u>Total</u>
21	<u>STRUCTURES AND IMPROVEMENTS</u>	14,624	13,122	17,761
211	Ground Improvements	1,830		
212	Buildings	3,900		
	Turbine Building	3,005		
	Administration Bldg. (Shop, Control Room, etc.)	885		
	Gate House	10		
218	Stacks	255		
219	Reactor Buildings	8,639		
	Containment Vessel	1,474		
	Reactor Building	3,938		
	Steam Gen. Bldg. and Reactor Aux. Bldg.	3,227		
22	<u>REACTOR PLANT EQUIPMENT*</u>	76,054	59,438	97,395
221	Reactor Equipment	10,906	8,523	13,966
221.1	Reactor Vessel	5,305	4,146	6,793
221.2	Reactor Controls	877	685	1,123
221.3	Reactor Shielding	1,236	966	1,583
221.4	Reactor Auxiliary Cooling	763	596	977
221.6	Reflector Elements	599	468	767
221.7	Cranes, Hoists, Handling Equipment	2,126	1,662	2,723
222	Heat Transfer Systems	39,934	31,209	51,140
222.1	Reactor Coolant System	6,360	4,971	8,145
222.11	Primary Pumps	4,476	3,498	5,732
222.12	Primary Tank	1,884	1,473	2,413
222.2	Secondary Coolant System	12,549	9,807	16,070
222.21	Pumps and Drives	2,883	2,253	3,692
222.22	Piping and Installation	3,853	3,011	4,934
222.23	Intermediate Heat Exchanger	5,813	4,543	7,444

* Includes Reactor Manufacturer Engineering, G&A, and Fees

TABLE 1.5 (continued)

		Expected Values		10% Probability		90% Probability	
		<u>Total</u>		<u>Total</u>		<u>Total</u>	
22	Reactor Plant Equipment (Continued)						
	222.3 Steam Generators	16,298		12,737		20,872	
	222.31 Steam Generators	8,933		6,985		11,446	
	222.32 Superheaters	3,543		2,773		4,544	
	222.35 Reheaters	3,812		2,979		4,882	
	222.4 Coolant Treatment and Inert Gas	4,727		3,694		6,053	
	Coolant Supply and Nitrogen System	2,020		1,579		2,587	
	Breathing Air and Argon System	2,707		2,115		3,466	
223	Nuclear Fuel Handling and Storage	3,399		2,657		4,353	
225	Radioactive Waste Treatment and Disposal	673		526		862	
226	Instrumentation and Control	13,312		10,404		17,047	
	226.1 Reactor Plant Control	8,496		6,640		10,880	
	226.2 Heat Transfer System Control	2,124		1,660		2,720	
	226.5 Radiation Monitoring	575		449		736	
	226.6 Steam System Control	879		687		1,126	
	226.6 Steam System Control	1,238		968		1,585	
227	Feedwater Supply and Treatment	3,010		2,352		3,855	
228	Steam, Condensate and Feedwater Piping	4,820		3,767		6,172	
23	<u>TURBINE GENERATING PLANT EQUIPMENT</u>		24,618		23,295		26,326
231	Turbine Generators	20,631		19,800		21,600	
	231.1 Foundation and Installation	998		800		1,200	
	231.2 Turbine Generators	19,633		19,000		20,400	
232	Circulating Water System	1,380		1,200		1,600	
233	Condensers and Auxiliaries	1,915		1,700		2,300	
234	Lube Oil System	75		65		90	
236	Turbine Plant Piping	585		500		700	
237	Aux. Equipment for Generators	32		30		36	
24	<u>ACCESSORY ELECTRICAL EQUIPMENT</u>		3,025		2,600		3,600

TABLE 1.5 (Continued)

	Expected Values	10% Probability	90% Probability
	<u>Total</u>	<u>Total</u>	<u>Total</u>
25 <u>MISC. POWER PLANT EQUIPMENT</u>	1,055	950	1,250
Allowance for Incomplete Design	<u>12,000</u>	<u>12,000</u>	<u>12,000</u>
TOTAL DIRECT CONSTRUCTION COST	131,376	111,405	158,332
Field Distributable Costs	13,441	13,020	15,151
Engineering, Inspection and Fees, Arch. - Engr	10,300	10,300	10,300
Contingency; General Uncertainty	<u>8,000</u>	<u>8,000</u>	<u>8,000</u>
Total—Less Customer Costs	163,117	142,725	191,783
Customer Costs at 20%	32,623	28,545	38,357
Sodium Inventory	<u>925</u>	<u>925</u>	<u>925</u>
TOTAL PLANT, 10 ³ \$	196,665	172,195	231,065
Net Plant Output, MWe	1,010	1,010	1,010
TOTAL PLANT, \$/kWe net	195	170	229

It should be noted that the costs shown in Tables 1.4 and 1.5 are based on 1968 dollars. A suggested guideline from Argonne National Laboratory was the use of 1967 dollars. For the early commercial LMFBR, the use of 1967 dollars would result in an estimated de-escalation of \$8,000,000, or a total plant of cost of \$187/kWe net.

1.6.2 Operating and Maintenance Costs (A. S. Gibson)

A summary of the annual maintenance and operating costs for the 1000 MWe LMFBR is given in Table 1.6. These costs can vary considerably from utility to utility due to different operating philosophies. On a conservative basis it is estimated that the subtotal of the costs, exclusive of insurance, shown in Table 1.6 could be 2 million dollars annually.

Table 1.7 shows the estimated annual labor costs and the staffing requirements for a single 1000 MWe LMFBR. The plant payroll costs shown in Table 1.6 were obtained from Table 1.7. Numerous different staffing possibilities exist depending upon the management

philosophy of the particular utility. Some utilities depend heavily upon contract maintenance for items such as the turbine, and particular instrumentation. Other utilities rely almost completely upon their own maintenance personnel. Moreover, some utilities rotate maintenance and technical personnel from unit to unit within their system depending upon the needs for their services. Other utilities assign personnel on a more or less full-time basis to a particular unit or plant. Table 1.7 is based upon a compromise between a philosophy which depends very heavily upon contract maintenance and technical assistance, and one which relies solely on utility personnel for these services. Some rotation of personnel is assumed.

Table 1.8 shows the annual cost of operating supplies and maintenance materials and services for the 1000 MWe LMFBR. Again the total costs in Table 1.8 were utilized in the generation of Table 1.6. Also, the items shown in this table reflect a modest amount of contract maintenance for particular items such as sodium pumps, the main plant computer, and other complex equipment.

TABLE 1.6

ANNUAL MAINTENANCE AND OPERATING COSTS FOR A 1000 MWe LMFBR (Costs in 1968 Dollars)

1.	Plant Payroll	\$ 669,000
2.	Operating and Maintenance Supplies and Contract Maintenance	699,000
3.	Utility Engineering and Supervision (5% of 2)	35,000
	Subtotal	\$1,403,000
	Utility General and Administrative Expense (at 20%)	208,000
	Subtotal	\$1,683,000
	Insurance (including nuclear)	~ 1,000,000
	Total	\$2,683,000
	Total, mills/kWh*	0.386

* At 80% capacity factor.

TABLE 1.7
ANNUAL LABOR COST OF 1000 MWe LMFBR

	Personnel Required	Annual Cost, \$
<u>Management</u>		
Superintendent	1	18,000
Assistant Superintendent	1	14,000
Office Manager-Storekeeper	1	9,000
Secretary	1	6,000
Clerk-typist	2	11,000
	6	58,000
<u>Technical</u>		
Power Plant Engineer (SOL)*	1	15,800
Nuclear Engineer (SOL)	1	14,900
Instrument Engineer	1	13,000
Chemical and Radiation Engineer	1	14,000
Technicians	4	50,000
Apprentice Technicians	2	19,000
	10	126,700
<u>Operations</u>		
Supervisor of Operations (SOL)	1	15,800
Shift Foremen (SOL)	5	77,000
Senior Control Operators (L)*	5	55,000
Control Operators (L)	5	50,000
Auxiliary Operators	4	36,000
Shift Helpers	4	29,000
Plant Guards	5	24,000
	29	286,800
<u>Maintenance**</u>		
Supervisor of Maintenance	1/3	15,800/3 5,200
Electrical Foreman	1/3	13,000/3 4,300
Mechanical Foreman	1/3	13,000/3 4,300
Electricians	2/3	20,000/3 6,700
Apprentice Electricians	2/3	16,000/3 5,300
Machinists	4/3	40,000/3 13,300
Apprentice Machinists	2/3	16,000/3 5,300
Certified Welders	2/3	20,000/3 6,800
Rigger	1/3	10,000/3 3,300
Tool Clerk	1/3	8,000/3 2,700
Machinist and Electrical Helpers	3/3	21,000/3 7,000
	20/3	192,800/3 64,200
Total Plant Direct Payroll		45 Full-Time personnel 535,700
		20 Pooled personnel
Fringe Benefits at 20%		107,100
Allowance for Overtime at 5%		26,000
Total Payroll Including Overtime and Fringe		668,800

* SOL = Senior Operator License
L = Operator License

** Maintenance personnel include total complement needed to service three 1000 MWe units.

TABLE 1.8
ANNUAL COST OF OPERATING SUPPLIES AND
MAINTENANCE MATERIALS AND SERVICES
ON A 1000 MWe LMFBR

1. Pumps: replacement parts and contract maintenance	\$ 55,000
2. Steam generators and IHX parts and maintenance	39,000
3. Shim and scram rod and reflector replacement	342,000
4. Instrumentation sensor replacement and repair	17,000
5. Replacement sodium and auxiliary sodium system parts	16,000
6. Control and safety equipment parts and consumables	22,000
7. Main computer contract servicing	75,000
8. Shield plug seals and refueling cell equipment	8,000
9. Gaseous system parts and consumables	25,000
10. Turbine system parts and miscellaneous replacement parts	100,000
	<u>\$ 600,000</u>

It is observed that the unit energy cost for maintenance and operating expense is shown as 0.386 mills/kWh in Table 1.6 and this energy cost is based on an 80% plant capacity factor. Argonne National Laboratory specified that the plant capacity factor should be defined equivalent to 0.9 times the availability factor. Studies performed elsewhere and reported in the Task III document indicate that if suitable component development is affected, a plant availability due solely to component reliability could be expected to be in excess of 95%.

An appropriate goal for plant availability based solely on instrumentation system reliability would also be in excess of 95%. Thus approximately two weeks per year outage each could be expected from both instrumentation and component causes. Additionally, a refueling study reported in this document indicates a refueling schedule of one week. Studies of water reactor operating experience demonstrate that one to two weeks of inspection time must be allowed during plant downtime. Assuming that some of these periods of downtime could be concurrent, an allowance of five weeks per year for plant outage is made.

This corresponds to a plant availability factor of 0.9. Thus the plant capacity factor is 0.9×0.9 or approximately 0.8. This capacity factor was utilized in all economic studies throughout the Task II and Task III work.

1.6.3 Fuel Cycle Costs (W. A. Sangster)

The fuel cycle costs of the reference final core design are presented herein in sufficient detail to allow any single item (such as unit cost of fabrication, fixed charge rate, shipping charges, etc.) or group of items to be readily identified. During the Task I and Task III efforts such detailed fuel cycle costs were not presented due to the large number of core concepts and fuel designs being evaluated.

Two sets of economic conditions were included in the Task I and Task III efforts. These were meant to encompass the economic uncertainties of fast reactor fuel cycle costs in the early 1980's. As such they served to put upper and lower bounds on the fuel cycle costs due to economic uncertainties only. The highest costs resulted from a set of assumptions that were

termed the "conservative" cost basis; this included a high fixed charge rate and high costs for fuel fabrication and recovery. The low fuel cycle costs were calculated from an "optimistic" cost basis and it was intended that these costs would be achievable only after the level of the fast reactor fuel business reached a sizeable throughput (~200 tonnes of U + Pu annually) so as to reduce the fabrication and recovery costs.

For the final Task II reference 1000 MWe LMFBF core design the fuel cycle costs were calculated according to an intermediate cost basis in addition to the above two bases used exclusively throughout Task I. This is labeled the "reference cost basis."

It is the reference cost basis fuel cycle costs that are presented in detail herein and the following is a summary of results for all three bases for the final 1000 MWe LMFBF core design:

	Core Fabri- cation Unit Cost	Recovery Package Unit Cost	Fixed Charge	Fuel Cycle Costs
	\$/kg (U+ Pu)	\$/kg (U+ Pu)	%/yr	mills/ kWeh
Optimistic Cost Basis	120	60	10	0.291
Reference Cost Basis	169.50	68.5	10	0.378
Conservative Cost Basis	280	120	10	0.746

These unit costs do not include any losses or financing costs.

The reference cost basis fuel cycle cost details are shown in Table 1.9. The method of these calculations differs from the one used in Tasks I and III in that all cost items are calculated in terms of \$/kg (U+ Pu) and only after the summation of these specific costs for each zone are the totals converted into mills/kWeh. This method, and the detailed itemization evident in Table 1.9 is based upon the method that is used to illustrate water fuel cycle costs to prospective utility customers

and this in-turn is based upon private ownership of nuclear fuel.

Table 1.9 contains three major parts. Part I is a summary of the reactor data by zone and includes most of the fuel design characteristics which are required for the fabrication plant cost model. Part II presents the numerous items in the reference fuel cycle cost calculation. Part III gives all the equations required to arrive at the various costs except for the fabrication cost model which is currently considered to be Company private.

The pertinent design data in Section part I of Table 1.9 is shown for each of the separate zones to be costed. The nuclear performance of the reactor is presented in Section 2.2.3.1 of this report and the eight distinct zones evaluated are identical to those indicated for an equilibrium fuel cycle in Table 2.5. Thus separate costing is performed for three zones of core fuel, three corresponding (physically attached) axial blanket zones and each of two rows of fuel subassemblies in the radial blanket.

The burnup shown for each zone is the average exposure when discharged from the reactor, and the power is also a time averaged value. The total mass of fissile and fertile material in each zone and the amount replaced at each refueling is shown; the replacement fuel is either one half, one third, or one fourth of the total. The amount of fissile material loaded and discharged from each zone at each refueling as shown in the table do not include any fabrication or reprocessing losses. Note that for the purpose of fuel cycle cost calculations only Pu-239 and Pu-241 are treated as fissile material. Thus the fissile masses in Table 1.9 do not exactly match the values shown in Table 2.5 because the latter also include the small amounts of U-235 present in depleted uranium. The U-235 has no intrinsic value and constitutes less than one half percent of the total fissile material in the reactor.

TABLE 1.9

FUEL CYCLE COST DATA AND DETAILS

	CORE ZONES			AXIAL BLANKETS			RADIAL BLANKET		UNITS	COMMENTS
	1	2(a)	2(b)	1	2(a)	2(b)	1(a)	1(b)		
I. REACTOR DATA										
1. Burnup	101265	98614	99523	9146	5784	6342	8664	4931	MWd/Te	[All data applies to an equilibrium fuel cycle and thermal reactor plutonium. (See Section 2.2.3.1)]
2. Power	1166.9	568.1	382.3	118.1	37.3	27.3	79.1	37.0	MW(t)	
3. Mass	6.729	3.365	3.365	7.540	3.770	3.770	8.000	8.767	Te (U + Pu)	
4. Refuel Batch	3.365	1.682	1.122	3.770	1.885	1.257	2.667	2.192	Te (U + Pu)	Total fuel in reactor
5. Fissile Refuel Loading	358.6	250.3	166.9	0	0	0	0	0	kg Pu(239 & 241)	Amount loaded and discharged every 365 days
6. Fissile Refuel Discharge	357.0	206.1	137.1	136.4	52.6	37.4	89.2	59.8	kg Pu(239 & 241)	Does not include small (~30 kg) amounts of U-235
7. Initial Enrichment	10.66	14.88	14.88	0	0	0	0	0	percent	Thus "gain" (before losses) almost 300 kg Pu(239 & 241)
8. Cladding O.D.	0.250	0.250	0.250	0.250	0.250	0.250	0.422	0.422	inch	Ratio of items 5/4 (Final enrichment = 6/4)
9. Cladding Thickness	10	10	10	10	10	10	15	15	mils	Nominal hot dimensions
10. Fuel Length	30	30	30	30	30	30	60	60	inch	Stainless steel
11. Vent Device Length	21	21	21	-	-	-	21	21	inch	Upper and lower axial blankets are each 15 inches long
12. Total Fuel Element Length	6.85	6.85	6.85	6.85	6.85	6.85	6.85	6.85	feet	Over-all including end pieces
13. Fuel Smeared Density	0.85	0.85	0.85	0.95	0.95	0.95	0.95	0.95	Fraction of Theoretical	Also "hot"; hot theoretical density of oxide = 10.5 gm/cc

TABLE 1.9 (Continued)

	CORE ZONES			AXIAL BLANKETS			RADIAL BLANKET		UNITS	COMMENTS
	1	2(a)	2(b)	1	2(a)	2(b)	1(a)	1(b)		Reference Cost Basis
II. <u>FUEL CYCLE COST RESULTS</u>										
A. <u>FRESH FUEL SUPPLY</u>										Value when loaded in reactor
1. Plutonium	1065.8	1487.8	1487.8	0	0	0	0	0	\$/kg (U + Pu)	Based on 10 \$/gm 239 & 241
2. Fabrication	169.5	169.5	169.5	40.7	40.7	40.7	35.9	35.9	\$/kg (U + Pu)	Delivered, includes cost of depleted uranium
3. Related Costs	0									
a. Losses	10.7	14.9	14.9	0	0	0	0	0	\$/kg (U + Pu)	Value of 1% losses during fabrication
b. Financing	22.2	31.0	31.0	0	0	0	0	0	\$/kg (U + Pu)	2.5 months on fuel material being processed prior to reactor loading
4. Total Value, Initial	1268.2	1703.2	1703.2	40.7	40.7	40.7	35.9	35.9	\$/kg (U + Pu)	
B. <u>Spent Fuel Credit</u>										Net credit discharged from reactor
1. Plutonium	1061.0	1225.1	1222.4	361.8	279.0	297.6	334.4	272.8	\$/kg (U + Pu)	Based on 10 \$/gm 239 & 241
2. Recovery Package										
a. Reprocessing	48.5	48.5	48.5	48.5	48.5	48.5	48.5	48.5	\$/kg (U + Pu)	30,000 \$/day, 67% U-235 Limit
b. Shipping	20.0	20.0	20.0	20.0	20.0	20.0	20.0	20.0	\$/kg (U + Pu)	As specified by ANL
3. Losses	10.6	12.2	12.2	3.6	2.8	3.0	3.3	2.7	\$/kg (U + Pu)	Value of 1% losses
4. Net Value, Final	981.9	1144.4	1141.7	289.7	207.7	226.1	272.6	211.6	\$/kg (U + Pu)	Note: \$/kg is based on kg (U + Pu) as loaded

TABLE 1.9 (Continued)

	CORE ZONES			AXIAL BLANKETS			RADIAL BLANKET		UNITS	COMMENTS
	1	2(a)	2(b)	1	2(a)	2(b)	1(a)	1(b)		
II. FUEL CYCLE COST RESULTS (continued)										
C. <u>Financing Costs</u>										
1. Pre-irradiation										Pre-irradiation financing is covered in Item A.3. b
2. Irradiation Period	2.0	2.0	3.0	2.0	2.0	3.0	3.0	4.0	years	Time in reactor, including 80% capacity factor
3. Irradiation Charge	220.5	284.8	426.7	33.0	24.8	40.0	46.3	49.5	\$/kg (U + Pu)	Based on 10%/year fixed charge rate (as specified by ANL)
4. Post-irradiation	49.1	57.2	57.1	14.5	10.4	11.3	13.6	10.6	\$/kg (U + Pu)	Assumes 0.5 year from discharge through recovery
5. Total Cost of Financing	269.6	342.0	483.8	47.5	35.2	51.3	59.9	60.1	\$/kg (U + Pu)	
D. Total Fuel Cycle Cost	555.9	900.8	1045.3	-201.5	-131.8	-134.1	-176.8	-115.6	\$/kg (U + Pu)	Reactor Total
E. Heat Cost	3.24	2.62	2.03	-1.31	-0.43	-0.29	-0.82	-0.44	¢/MBtu	4.60
F. Electrical Energy Cost	0.267	0.215	0.166	-0.108	-0.035	-0.024	-0.067	-0.036	mills/kWeh	0.378

TABLE 1.9 (Continued)

III. FUEL CYCLE COST EQUATIONS

The following equations apply to all zones; items in parentheses refer to items in the units as given in the Table.

		UNITS	COMMENTS
A. Fresh Fuel Supply			
1. Plutonium			
(A. 1)	$= \frac{10 \times (\text{Fissile Refuel Loading})}{(\text{Refuel Batch})}$	$\frac{\$}{\text{gm (239 \& 241)}} \frac{\text{Kg (239 \& 241)}}{\text{Te (U + Pu)}} \left[\frac{10^3 \text{ gm Te}}{\text{kg} \times 10^3 \text{ kg}} \right] = \frac{\$}{\text{kg (U + Pu)}}$	10\$/gm is the reference plutonium value specified by ANL
2. Fabrication			
(A. 2)	The equations for fuel fabrication are based upon a large mixed oxide fabrication plant of GE design. This plant design is currently considered to be Company private	$\frac{\$}{\text{kg (U + Pu)}}$	Costs are thus "adjusted for design."
3. Related Costs			
a. Losses	(A. 3. a) = 0.01 × (A. 1)	$\frac{\$}{\text{kg (U + Pu)}}$	1% losses are specified by ANL.
b. Financing	(A. 3. b) = $0.10 \times \frac{2.5}{12} \times (\text{A. 1})$	$\frac{1}{\text{year}} \frac{\text{month} \times \text{year}}{\text{month}} \frac{\$}{\text{kg (U + Pu)}} = \frac{\$}{\text{kg (U + Pu)}}$	10% per year is the reference "interest" or fixed charge rate specified by ANL. (Sometimes a lower rate like 5% is charged here.)
4. Total Value, Initial			
(A. 4)	= (A. 1) + (A. 2) + (A. 3. a) + (A. 3. b)	$\frac{\$}{\text{kg (U + Pu)}}$	
B. Spent Fuel Credit			
1. Plutonium			
(B.1)	$= \frac{10 \times (\text{Fissile Refuel Discharge})}{(\text{Refuel Batch})}$	(Same units as Item A. 1) $\frac{\$}{\text{kg (U + Pu)}}$	
2. Recovery Package			
a. Reprocessing	(B. 2. a) { Both of these charges were specified by ANL; the average discharge enrichment of this reactor is found to be 0.0612 and thus the throughput to the reprocessing plant is 619 kg per day, hence 30000/619 = 48.5 \$/kg for reprocessing.	$\frac{\$}{\text{kg (U + Pu)}}$	
b. Shipping	(B. 2. b)	$\frac{\$}{\text{kg (U + Pu)}}$	

TABLE 1.9 (Continued)

III. FUEL CYCLE COST EQUATIONS (Continued)

		UNITS		COMMENTS
3. Losses	(B. 3) = 0.01 × (B. 1)		$\frac{\$}{\text{kg (U + Pu)}}$	1% losses are specified by ANL (BWR practice assumes 2% losses here)
4. Net Value, Final	(B. 4) = (B. 1) - (B. 2.a) - (B. 3)		$\frac{\$}{\text{kg (U + Pu)}}$	
C. <u>Financing Costs</u>				
2. Irradiation Period				
(C. 2)	$= \frac{(\text{Burnup}) \times (\text{Mass})}{0.8 \times 365 \times (\text{Power})}$	$\frac{\text{MWd}}{\text{Te}} \times \frac{\text{year}}{\text{d}} \times \frac{\text{Te}}{\text{MW}}$	= year	The plant capacity factor is 0.8
3. Irradiation Charge				
(C. 3)	$= \frac{(A. 4) + (B. 4)}{2} \times 0.10 \times (C. 2)$	$\frac{\$}{\text{kg (U + Pu)}} \times \frac{1}{\text{year}} \times \frac{\text{year}}{1}$	= $\frac{\$}{\text{kg (U + Pu)}}$	All "depreciations" are linear. 10% per year is the fixed charge rate specified by ANL
4. Post-irradiation				
(C. 4)	$= (B. 4) \times 0.10 \times 0.5$	(as in C. 3 above)	= $\frac{\$}{\text{kg (U + Pu)}}$	Assumes one half a year from discharge through recovery
5. Total Cost of Financing				
(C. 5)	$= (C. 3) + (C. 4)$		$\frac{\$}{\text{kg (U + Pu)}}$	
D. <u>Total Fuel Cycle Cost</u>				
(D)	$= (A. 4) - (B. 4) + (C. 5)$		$\frac{\$}{\text{kg (U + Pu)}}$	The kg (U + Pu) still refers to each zone, thus these costs cannot be added without adjustment for each zone's relative throughput
E. <u>Heat Cost</u>	The above Fuel Cycle Costs are converted to heat costs for each zone by relating to the burnup and power fraction of each zone. The 'total power' is simply the sum of the various zone powers.			
(E)	$= \frac{(D) \times (\text{Power}) \times 100 \times 10^6}{(\text{Burnup}) (\text{Total Power}) \times 3413 \times 24}$	$\frac{\$}{\text{kg}} \times \frac{\text{kg}}{\text{kWd}} \times \frac{\text{¢}}{\$} \times \frac{\text{kWh}}{\text{Btu}} \times \frac{\text{d}}{\text{h}} \times \frac{\text{Btu}}{\text{MBtu}}$	= $\frac{\text{¢}}{\text{MBtu}}$	Note: $\frac{\text{MWD}}{\text{Te}} = \frac{\text{kWd}}{\text{kg}}$
F. <u>Electrical Energy Cost</u>				
(F)	$= \frac{(E) \times 8161 \times 10}{10^6}$	$\frac{\text{¢}}{\text{MBtu}} \times \frac{\text{Btu}}{\text{kWh}} \times \frac{\text{MBtu}}{\text{Btu}} \times \frac{\text{mills}}{\text{¢}}$	= $\frac{\text{mills}}{\text{kWh}}$	8161 Btu/kWh is the net cycle heat rate

The equations used to calculate the specific costs in part II of Table 1.9 are all covered in part III except for the proprietary fabrication plant model. A simplified flow diagram of the model is included here in Figure 1.4 and this summarizes the items considered in the fabrication cost of 169.5 \$/kg (U + Pu) for the reference core design. Note that the depleted uranium for the core is purchased as uranyl nitrate and its assumed cost of 3 \$/kg U is included in the 169.5 \$/kg (U + Pu) for core fuel fabrication. The depleted uranium for the blankets is assumed to be purchased as ceramic grade UO_2 and a cost of 5 \$/kg U for this is included in the 40.7 and 35.9 \$/kg (U + Pu) charged for the axial and radial blankets respectively. The fissile feed is plutonyl nitrate from the reprocessing plant but the cost of plutonium is treated as a separate item as shown in Table 1.9.

The specific (\$/kg (U + Pu)) fuel cycle cost totals for each zone (item D in the table) are first converted to unit heat costs by the fraction of power and average burnup of each zone. The total of all eight zones is thus the reference heat cost for the whole reactor and this is seen to be 4.60 cents per million Btu. Following this, the electrical energy costs are readily obtained from the net plant heat rate. The total net electrical energy cost for the reference design is 0.378 mills per kilowatt hour. This value is the reference fuel cycle cost of the final 1000 MWe LMFBR design based upon an equilibrium fuel cycle fed with thermal water reactor plutonium. The results of a fast-recycle plutonium fuel cycle is covered in Task III and not reported herein.

The effect of the value of plutonium on the fuel cycle cost is shown in Figure 1.5 for all three cost bases. The principal cost inputs to the optimistic and conservative cost bases are as follows:

	<u>Optimistic</u>	<u>Conservative</u>
<u>Fabrication Cost</u>		
\$/kg (U & Pu)		
(Item A.2 in Table 1.9)		
Core Zones	120	280
Axial Blankets	30	70
Radial Blanket	24	56
<u>Recovery Package</u>		
\$/kg (U & Pu)		
(Items B.2.a and B.2.b)		
Core and Axial Blankets	60	120
Radial Blankets	50	100
<u>Fixed Charge Rate</u>		
%/year		
	10	13

With the above changes to the input the optimistic and conservative cost bases are calculated in the same manner as the reference costs in Table 1.9. Note that at a fixed charge rate of 10 percent per year the fuel cycle costs are fairly insensitive to the value of plutonium for this particular design. This can be seen in Figure 1.5; and it is the increase to 13 percent per year that causes the conservative cost basis to have an increasing fuel cycle cost versus plutonium cost trend.

The fuel cycle costs for the reference cost basis were also calculated by the method used previously for Tasks I and III and the results are shown in Table 1.10. In this table, the various items in the fuel cycle cost are grouped as nearly as possible to the components defined by ANL. The assignment of the various items from Table 1.9 to the cost components in Table 1.10 is summarized on the latter to avoid confusion.

FIGURE 1.4 P-O₂-UO₂ ROD FUEL FABRICATION PROCESSES

FIGURE 1.5 FUEL CYCLE COSTS FOR REFERENCE CORE DESIGN VERSUS VALUE OF FISSILE PLUTONIUM

TABLE 1.10
REFERENCE FUEL CYCLE COSTS
(mills/kWeh)

	Core	Axial Blanket	Radial Blanket	Total
Fabrication	0.148	0.040	0.025	0.213
Recovery	0.060	0.067	0.040	0.167
"Pu Credit"	0.128	-0.318	-0.20	-0.399
"Inventory"	0.312	0.044	0.041	0.397
Total	0.648	-0.167	-0.103	0.378

Component	Items Included (Reference Table 1.9)
Fabrication	(A.2)
Recovery	(B.2.a) + (B.2.b)
"Pu Credit"	(A.1) - (B.1) + (A.3.a) + (B.3)
"Inventory"	(A.3.b) + (C.5)

SECTION II

REACTOR SYSTEM

2.1 INTRODUCTION

2.1.1 System Functions

The principal functions of the reactor system are as follows:

1. Provide a complex of fissionable materials, structural materials, control materials and coolant arranged so that core and blankets can operate economically and safely as a breeder reactor and transfer heat to the coolant.
2. Utilize instrumentation and controls which detect the onset of abnormal conditions sufficiently early to take automatic corrective action and prevent excessive damage to the reactor or plant.
3. Provide interaction of the reactor controls with the plant control system so that the reactor power change automatically responds to changes in power demand.
4. Provide fuel handling equipment so that the used fuel may be removed from the reactor and replaced with new fuel.
5. Provide means for repairing portions of the reactor if unforeseen difficulties develop after construction and operation.
6. Include sufficient shielding to limit biological doses to acceptable levels and to limit activation of and neutron damage to components of the primary heat transfer system.

2.1.2 Summary Description

The reactor system consists of the following equipment:

1. Core support structure,
2. Fuel bundles,
3. Reflector bundles,
4. Control rods and drives,
5. Reactor instrumentation,
6. Reactor shield plug,
7. Refueling cell, and
8. Refueling equipment.

Each of these items is discussed below in the detailed description. A summary description follows.

The core support structure is a stainless steel cylindrical shell which contains the peripheral support rings for the reflector bundles and the lower grid structure. Included on the lower grid structure are fuel bundle supports which contain venturi meters for measuring coolant flow through each fuel channel. Attached to the lower grid structure is a perforated bottom plate which admits coolant to the flow channels but prevents large debris from passing into or blocking the fuel channels. The entire core support structure is assembled and installed as a single unit. In an emergency, it could be removed since it is bolted to the reactor vessel. (This is discussed under maintenance below.)

The fuel bundles consist of uranium-plutonium oxide fuel in 1/4-inch o. d. stainless steel pins, arranged on a triangular pitch in bundles which are bounded by hexagonal stainless steel channel walls. The active region of the fuel constitutes a 30-inch-long enriched portion which is located between a lower and upper blanket of depleted uranium. The axial

blankets and enriched fuel are in continuous tubes of stainless steel. The upper portion of these pins includes a plenum which allows venting of fission products but prevents sodium coolant from entering into the fuel regions. The fuel pins are separated by entwined wire spacers, and are supported as an assembly by locking pins at the bottom and by the channel walls at the periphery of the bundles. The fuel bundles rest on the core support structure in a closely packed array. Around the outside of this array, radial blanket fuel bundles of depleted uranium oxide are located. A row of reflector elements is located outside of the radial blanket. The radial blanket and reflector serve to absorb or reflect neutrons so as to utilize them in the most efficient manner and minimize activation or structural damage to parts of the primary heat transfer system. All of the fuel bundles are designed to resist the "lifting" pressure of the inlet coolant by a hydraulic hold-down device. A back-up mechanical hold-down is provided from the shield plug structure. During operation of the reactor, the fuel channels deflect under internal pressure until they touch adjacent channels. In this manner, core "clamping" is achieved. Individual orificing is provided in each fuel channel to assure proper distribution of coolant flow.

Two types of control rods are provided: shim control and scram. Sufficient scram rods are included for normal shutdown plus an independent back-up shutdown. The scram rods are constructed of boron carbide and clad with stainless steel. They are normally withdrawn into the axial blanket region of the core when the reactor is operating. The shim rods are constructed of tantalum and may be partially inserted in the core when the reactor is operating.

All of the control rods pass through modified fuel bundles which are partially filled with fuel pins. A channel is provided to allow smooth movement of the rods through the modified fuel bundle. For fuel transfer, the rods are disconnected above the core, and the drive mechan-

isms are removed with the shield plug. The absorber portion of the rod is handled as an integral part of its associated fuel bundle.

The drives for the shim rods are electric motors with a rack and pinion gear arrangement. The scram drives are pneumatic and can only be lifted from the core one at a time. All of the rod drives are sealed units operating in argon from the reactor cover gas atmosphere which has been cleaned of entrained sodium vapor. The argon is continuously injected at the top of the drive mechanisms from which it flows downward into the reactor cover gas region. A description of the inert gas system is presented in Section 3.8.

Reactor instrumentation includes neutron flux monitors located in the blanket or reflector region of the reactor, independent flow measurements of each fuel channel, and temperature measurements at the top of the fuel channels. The temperature measurements are made by thermocouple probes which protrude from the top shield structure to a position above the outlet of the fuel channels. Gross measurement of the sodium coolant for delayed neutrons or specific fission products may also be performed by sampling sodium in the outlet plenum above the core.

The reactor instrumentation and the control rods and drives are functionally arranged with the safety circuits to require set-backs or scram upon receipt of signals which indicate an incipient unsafe condition. The reactor instrumentation and control rods and drives are also interconnected with the plant safety system so that automatic control of the plant power output can be made in response to system demand. This plant control system is described in Section 3.13.

The reactor shield plug is constructed of iron-magnetite concrete and rests over the top of the reactor on the shield deck. Locking lugs securely seal it in place to the shield deck. It is lifted from its position over the re-

actor by removing the locking lugs and activating two lifting screws on each side of the plug. These screws are secured to the side walls of the refueling cell and permit raising the shield plug to the top of the cell so that refueling mechanisms may be used. The shield plug has control rod drives and instrument thimbles mounted to its upper surface. These drives and thimbles penetrate the plug to a region above the core. Just below the lower face of the shield, a series of horizontal steel plates are mounted which have loosely fitting seals through which the instrument thimbles and control rods penetrate. During reactor operation, cleaned argon gas continuously flows through these seals to prevent condensation of sodium vapor on colder surfaces. This is discussed in the primary heat transfer system description (Section 3.1).

The refueling cell is a steel-lined concrete structure over the reactor vessel. The cell is filled with an inert gas. Provision is made so that personnel can enter the cell for routine maintenance when the shield plug is in place and the reactor is shut down. During this routine maintenance procedure, the personnel wear protective suits with separate air systems similar to those provided for the SEFOR reactor. With the reactor shield plug removed, fuel can be removed from the reactor by the fuel shuffling and transfer mechanisms, and stored in the decay storage pit in the refueling cell. New fuel is placed into the reactor by the same shuffling and transfer mechanisms. Since the reactor utilizes vented fuel which vents

radioactive krypton and xenon to the cover gas region, it is necessary to purge this radioactive gas region with clean argon before lifting the shield plug for refueling. The purge and discharge of the radioactive gases are described in the inert gas system description.

If an unforeseen problem should develop in the reactor which requires unusual maintenance procedures or equipment, this equipment can be brought into the refueling cell, and the reactor plug removed for access to the problem area. The special tools can then be used for direct access to the core. If necessary, sodium level can be dropped to assist in this operation. In extreme cases, temporary sodium storage could be provided, and complete access to the core would be possible.

2.1.3 System Design Requirements

The need for design requirements of the reactor system is derived entirely from the arbitrary nature of certain design parameters, or the previous optimization of these parameters which precludes the need for further investigation. If a complete optimization of the reactor were being performed, it would be sufficient to specify that the design requirement of the reactor should be to provide some fixed rate of heat production under safe and economically optimum conditions. Since complete optimization is never possible, some "design requirements" were needed in the Task II and Task III work. These requirements are the following:

<u>Item</u>	<u>Requirement</u>
1. Fuel type	Mixed uranium-plutonium oxide with gas bond
2. Cladding and core structural materials	Stainless steel
3. Core thermal output	Sufficient to produce in the range of 1000 to 1100 MWe net
4. Cladding strain criterion	Maximum allowable for fluence and operating temperature

<u>Item</u>	<u>Requirement</u>
5. Maximum fuel temperature	5200° F
6. Fuel pin array	Triangular
7. Size of fuel bundles	Fixed by considerations of fuel handling time and safety considerations.
8. Fuel pin design	Vented or sealed pin depending upon optimum economics
9. Fuel burnup	Fixed by optimum economic conditions
10. Fuel channel design	Non-free standing type
11. Minimum fuel cladding thickness	Fixed by limits of allowable creep, stress, or manufacturing and corrosion allowances
12. Fuel density and specific power	Fixed by optimum economic conditions
13. Maximum mixed mean coolant temperature outlet	1150° F
14. Core outlet coolant temperature deviation from nominal	$\pm 20^{\circ}$ F throughout control range of 20% load to 100% load
15. Core inlet coolant temperature deviation	$\pm 20^{\circ}$ F throughout control range of 20% load to 100% load
16. Rate of change of fuel temperature	Fixed by plant control system design requirements
17. Application of core peaking factors	Multiplication of all factors
18. Reactor geometry	Fixed by optimum economic conditions
19. Core coolant temperature rise and flow rate	Fixed by optimization of core and plant parameters
20. Core orificing	Orificing on discrete rows of fuel channels; fixed position throughout core life
21. Core cover gas	Argon
22. Core hold-down	Hydraulic with mechanical back-up

<u>Item</u>	<u>Requirement</u>
23. Safety instrumentation and controls	Sufficient sensitivity to detect and limit damage propagation to seven fuel bundles or less
24. Total primary control rod worth	Reactivity change from burnup of one cycle, and from cold shutdown to full power, plus one stuck rod, plus void worth of one fuel bundle, plus melting and slumping of one fuel bundle, plus 1.50\$, plus 20% contingency.
25. Total backup control rod worth	Same as primary except stuck rod not included
26. Maximum worth of shim rod	1.10 \$ or less
27. Maximum worth of fuel bundle	1.10 \$ or less
28. Scram rod acceleration	1 g or better
29. Maximum fuel damage criterion	Gross fuel cladding failure in more than one fuel bundle
30. Control rod materials	Boron carbide or tantalum
31. Refueling schedule	1 year
32. Refueling method	Hot cell type
33. Refueling transfer mechanism cooling capacity	Sufficient cooling capacity for indefinite delay of transport of removal fuel in argon cell atmosphere
34. Refueling cell personnel availability	1 day after reactor shutdown
35. Thermocouple and control rod repair or replacement	Removable from upper shield plug
36. Core component repair	All core components removable in an emergency

2.2 DETAILED DESCRIPTION

2.2.1 Mechanical Design of Core and Fuel (J. H. Germer)

2.2.1.1 Fuel Pins

The design of the fuel pins is shown on Figures 2.1 and 2.2. Each pin is supported at its lower end and is subjected to upward sodium flow. It contains a 30-inch-long region of mixed UO_2 - PuO_2 pellet fuel between two 15-inch axial blanket regions of depleted UO_2 pellets.

NOTE:
FOR DIMENSIONS AND QUANTITIES
SUPERSEDING THIS DRAWING SEE
REACTOR CORE No 2-174 F140

FIGURE 2.1 FUEL BUNDLE-UPPER END (SIMILAR TO
GE DRAWING 277E170 SH. 1)

FIGURE 2.2 FUEL BUNDLE-LOWER END (SIMILAR TO
GE DRAWING 277E170 SH. 2)

The choice of oxide fuel meets the objectives of this study. When sufficient information becomes available some other fuel material may prove to be better. The Task III report presents a discussion of carbide fuel.

A 1/4-inch-long pellet of MgO is located at each end of the fuel and blanket column to separate the heat generating region from the upper vent region and from the lower end plug. The upper 22 inches of the fuel pin form a space for venting fission product gas to the coolant stream, thus preventing a buildup of internal gas pressure. The entire fuel pin is enclosed in a type 316 stainless steel cladding tube of 0.250-inch outside diameter and 0.010-inch wall thickness (0.248-inch diameter at room temperature). The choice of vented-to-coolant fuel meets the objectives of this study. Another means for handling the fission gases may eventually prove to be more attractive. The Task III report presents a discussion of sealed-pin fuel.

The vent is designed to provide a trap which prevents the entry of liquid sodium into the fuel region. The trap includes an inner cup surrounded by a thin annulus through which the gas rises. The gas then passes downward inside the cup, and leaves the pin through a small centrally located tube sealed through the upper end cap. When the fuel temperature is lowered, or the external pressure is raised, sodium flows into the inner cup. The size of the cup is such that the sodium level never reaches the top of the inner cup to spill over into the fuel. The vent performance is described in Section 2.2.3.3.

The inner cup also prevents the fuel pellets from moving upward during handling and shipping. The construction of the upper end of the inner cup provides for upward axial expansion of the fuel during operation. The edges of the cup are turned inward to form an interference fit with a downward extension of the top end plug. These cup edges will deform to permit a one-inch upward motion after a predetermined axial force is exceeded.

The upper and lower end plugs are welded to the fuel pin cladding tube to form a sealed unit, except for the small vent tube extending through the upper end plug. During fabrication, the vent tube is tightly sealed by a soft soldered pinch-off at the upper end. This permits handling the fuel pin and assembly without contamination of the fuel by air and without escape of plutonium oxide. When the fuel pin is immersed in the hot sodium, the soft solder will melt and open the vent tube.

The upper end plug is covered by a protective cap which is tack welded in place and prevents damage to the small vent tube. It also serves as an anchor for the upper end of the spacer wire. A small hole in the side of this protective cap permits escape of the vented gas.

A horizontal strip is welded to the lower end plugs of 11 to 21 fuel pins to form a single row across the fuel bundle. Each row of fuel pins is installed as a unit with the horizontal strip fitting into a slot in the lower support structure of the fuel bundle. The 21 rows of fuel pins are retained by two horizontal pins, as shown in Figure 2.1.

Helical wire spacers are wrapped around the fuel pins in order to control their spacing. The choice of helical wire spacers meets the objectives of this study, but another spacer design may eventually prove to be more attractive. A discussion of other designs is presented in the Task IV report.

These wire spacers are welded into holes in both the lower end plug and the upper protective cap of the fuel pin. The wires form a pitch of six inches per turn. One third of the fuel pins in the center part of the fuel bundle do not have spacer wires, and rely on the wires on the adjacent pins for their spacing. The fuel pins forming the outer row of the fuel bundle all are provided with wires, since spacing from the hexagonal channel wall is required. The fuel pins which have no wires

are made with a slightly longer upper protective cap (not shown in Figure 2.1) to permit the use of a "no-go" inspection gage to assure their proper location. If two fuel pins without wires were placed in adjacent locations they could touch and cause local burnout.

2.2.1.2 Fuel Bundles and Pin Supports

The fuel bundles are shown in Figures 2.1 and 2.2. Detailed dimensions in the cross section and the number of pins are shown in Figure 2.3. The dimensions on Figure 2.3 supersede the corresponding ones on Figure 2.1 and on Figure 2.2. Two hundred thirty-four of these fuel bundles, along with 31 modified fuel bundles with control rod guides, make up the reactor core. These two types of fuel bundles are similar except for the addition of a control rod guide tube which replaces all but the outer four rows of fuel pins in the control rod fuel bundle. A further description of the control rod fuel bundle is given under "Control Rods" (Section 2.2.1.6).

Each fuel bundle contains 331 fuel pins arranged in 21 rows (as discussed under "Fuel Pins," Section 2.2.1.1). The lower horizontal end strips which join each row of fuel pins fit into a pair of slotted projections in a fuel support casting. Two horizontal pins hold these end strips in place. The horizontal pins are retained by the hexagonal channel wall.

The fuel support casting is connected to the lower end fitting of the fuel bundle by six tension rods located near the corners of the hexagonal channel wall. It is also joined with a flexible fitting to an inverted cup containing a seal which fits over the fuel bundle support. A second seal, which fits over the fuel bundle support, is located near the lower end of the fuel bundle. Sodium flows from holes in the side of the fuel bundle support and enters the fuel bundle at an elevation between these two seals. A vent tube within the fuel bundle support maintains the pressure above the fuel bundle support at the same level as the pres-

sure surrounding the lower end of the fuel bundle, which in turn is vented to the fuel bundle outlet. The result is that there is no net upward hydraulic force on the fuel bundle due to the pressure drop. This pressure balance is not complete, since for practical assembly reasons it is desirable to have the upper seal slightly smaller in diameter than the lower seal. Also, leakage of sodium past the upper seal will raise slightly the pressure above the upper seal. This unbalanced force, however, can easily be kept to a value considerably less than the weight of the fuel bundle.

The two seals are constructed in such a way as to permit radial misalignment of the fuel bundle by 1/8 inch in any direction, and to permit angular misalignment. This is accomplished at the upper seal by rocking the inverted cup assembly.

The lower seal permits misalignment by sliding ring construction, where the small split seal ring is retained between two continuous rings. The sliding ring construction could not easily be used for the upper seal because the greater radial size would severely limit the flow area. Clearances between the fuel bundle support and both the sliding rings and inverted cup assembly are sufficiently large to assure reliable assembly and disassembly, but are close enough to avoid excessive flow if the split ring were to fracture. The split rings are installed in such a way that their failure will not cause them to become lost.

There is a two-inch axial length over which the two seals and the fuel bundle support are engaged. This prevents an upward pressure load even with the fuel bundle not entirely seated. A clearance above the fuel bundle of about one inch to a fixed backup hold-down structure assures that bundle movement will not exceed the engagement length of the seals.

The hexagonal fuel bundle channel wall is welded to both the upper and lower end fittings

REFLECTOR BUNDLE

CORE PLAN

SECTION A-A
ELEVATION

NOTE:
DIMENSIONS ARE AT 850°F
(70°F IN PARENTHESIS)
0.8% EXPANSION ASSUMED 70°F-850°F

FIGURE 2.3 FINAL FUEL AND CORE DESIGN
DIMENSIONS (SIMILAR TO GE DRAWING
174F140)

GEAP-5678

FCF-296X113

at its extreme ends only, thus avoiding stress concentrations. The lower end of the channel wall is formed with a transition to a circular shape in order to minimize the stress concentration at the weld due to internal pressure. At this point, the pressure differential is greatest. This construction requires that the channel wall be installed from the lower end of the fuel pin bundle, and that both end welds be made after assembly.

The upper end of the fuel bundle has a three-bladed handle attached to the upper fitting. This fitting is welded to the upper end of the hexagonal channel wall, and incorporates a "hold-down grid" to prevent a fuel pin from being pushed out of the assembly in the event of a circumferential cladding failure, or a failure of the lower end attachment.

2.2.1.3 Radial Blanket Bundles

The radial blanket bundles (shown in Figure 2.4) are similar to the fuel bundles, except that they are designed for considerably lower heat generation. There is a considerable pressure difference between the radial blanket bundle and the adjacent fuel bundle, due to the much smaller flow in the radial blanket bundle. For this reason the radial blanket bundle has a thicker (0.120-inch) hexagonal channel wall and no transition to a round shape at the bottom.

The pressure drop through the radial blanket bundle is much smaller than that through the fuel bundles. By allowing a large pressure drop in the fixed inlet structure, and by providing a small-diameter lower seal, it was possible to eliminate the pressure balanced design which was used in the fuel bundles. The inlet orifice is designed such that flow passes in through several radial holes after passing up through a thin annulus. This arrangement offers less probability of flow blockage than a single hole.

The fuel pin outside diameter in the

blanket bundle is 0.422 inch. There are 127 pins in each assembly. The low heat generation permits a smaller pitch-to-diameter ratio than in the fuel bundle. Spacer wires of much tighter pitch (2 inch) are fitted to all pins. This tight pitch tends to improve the flow mixing, which is particularly desirable adjacent to the core where the power gradient is very steep.

The handle of the radial blanket bundle is the same as that of the fuel bundles, and the overall bundle length is the same. This permits refueling with the same equipment. Details of the lower end preclude the accidental location of a radial blanket bundle in the core region.

Detailed dimensions in the cross section, and the number of pins are shown on Figure 2.3. The dimensions on Figure 2.4 are not those which were finally selected for the reference design.

2.2.1.4 Reflector Guide Bundles

The reflector guide bundles (shown on Figure 2.5) form the outer periphery of the reactor and surround both the core and the radial blanket. Their primary function is to provide rigidity to the edge of the reactor and to resist radial expansion of the core when the fuel bundles are pressurized. They also serve a lesser function of reflecting neutrons back into the reactor.

The reflector guide bundles are constructed primarily of solid stainless steel with a minimal amount of coolant channels to prevent overheating. Three such coolant channels are formed by the gap between four concentric cylinders. Uniformity of the gaps is provided by small welded projections on the outsides of the inner cylinders. The composite structure has a bending strength and stiffness essentially the same as if it was solid steel.

The lower ends of these reflector guide bundles are built without coolant passages

from the inlet plenum. Flow enters the bundles from the space above the reactor support plate and consists mainly of sodium which has leaked past the fuel bundle inlets. Sodium leaves side openings near the top of the bundle and passes radially outward into the radial shielding region, and finally exhausts at the lower end of the shield region into the reactor vessel. Flow from the reflector guide bundles bypasses the intermediate heat exchangers and permits this region to be considerably overcooled.

As with the fuel and with the radial blanket bundles, the dimensions shown in Figure 2.3 are the reference values. Those given on Figure 2.5 are not the reference values.

2.2.1.5 Core Structure

The reactor support plate and side structure are fabricated as a single unit. (see Figures 2.6 and 2.7). This unit incorporates most of the more critical machining which relates to the core structure. Its lower section is made up of two horizontal plates separated by a 19-inch gap. The gap contains a strong gusset structure which withstands both the upward pressure of the sodium flow and the downward weight of the core. The lower support plate is perforated with many small holes in order to prevent loose pieces of material from covering the entrances of the fuel bundles. The gussets between the two support plates are perforated to permit cross flow in the event that some of the holes in the lower plate become blocked.

The upper core support plate has support fixtures mounted on it for the fuel bundles, for the radial blanket bundles, and for the reflector guide bundles. These fixtures are bolted to the top of the support plate. The support fixtures are shown in more detail in the drawings of the particular bundles (Figure 2.2, Figure 2.4, and Figure 2.5). The bolted joints permit replacement of the support fixtures

without removal of the entire core structure. The bolt heads are of a special construction which allows them to be staked at three places to prevent loosening except by a torque which exceeds a predetermined value.

Flow holes in the sides of the fuel bundle supports are made in different sizes to provide variations in core orificing. The holes in the bundle supports at the center of the core are as shown in Figure 2.2, but are smaller at the outer part of the core.

The outer cylindrical shell of the core structure is welded to the two support plates and to three horizontal rings which are shaped on their inner surface to the contour of the outside of the reflector guide bundles. These rings, along with the outer shell, resist the radial expansion of the core due to internal pressure in the fuel bundles. The space between these rings contains steel shielding to minimize the fast neutron irradiation of the structure. At 24 locations, the rings are penetrated by guide tubes which serve as temporary storage locations for the bundle transfer thimbles during the refueling operation. The core structure is held down by 24 bolts located at the bottom of each guide tube.

The nominal clearance between all fuel bundles is 0.030 inch when the reactor and surrounding structure are at the same temperature. When the reactor is at operating condition, the surrounding structure controlling the radial guide bundles is near the inlet temperature (810°F), and the upper ends of the fuel bundles are at the outlet temperature (1150°F), causing a relative thermal expansion of about 0.019 inch. Thus the actual clearance between bundles at the outlet will be about 0.011 inch per bundle, and any further expansion will be suppressed.

In addition to this differential thermal expansion, the fuel bundles are internally pressurized to a degree which is strongly a function of orificing and location of the orifice (inlet, or outlet, or combination). This internal pressur-

FIGURE 2.4 RADIAL BLANKET BUNDLE (SIMILAR TO GE DRAWING 277E209)

GEAP-5678

NOTE:
1. MATERIAL, ALL PARTS, TO BE
STAINLESS STEEL
2. FOR DIMENSIONS AND QUANTITIES
SUPERSEDING THIS DRAWING SEE
REACTOR CORE No. 2-174 F140

FIGURE 2.5 REFLECTOR GUIDE BUNDLE (SIMILAR TO
GE DRAWING 174F112)

FIGURE 2.6 REACTOR-PLAN VIEW (SIMILAR TO GE
DRAWING 277E192 SH. 1)

FIGURE 2.7 REACTOR-ELEVATION (SIMILAR TO GE
DRAWING 277E192 SH. 2)

ization causes the bundles to expand and to touch each other over essentially their entire length, thus forming a rigid structure. By controlling the outer regions of the core with an orifice at the outlet end, these outer bundles will have a greater internal pressure than the other bundles. This causes the inner part of the core to be clamped tightly, such that the bundles in this region touch each other firmly. In this manner, the central part of the core will thermally expand in a reliable manner as outlet temperature is raised.

There has been no allowance in this design for volumetric expansion of channel wall material due to fast neutron irradiation. If further data indicate that expansion of the metal is as much as possible pessimistic extrapolation indicates, it may be necessary to modify the design.

Also incorporated in this core structure are provisions for inlet flow measurement. This is discussed in more detail in Section 2.2.4.3. The decision to measure flow at the inlet of each fuel bundle meets the objectives of this study. Outlet flow measurement may eventually prove to be more attractive. A discussion of outlet flow measurement is presented in the Task IV report.

Individual tubes connect to the inlet venturis of all of the fuel bundles, including those with control rods. These tubes are collected in six vertical cylindrical enclosures at the six corners of the reactor. Electrical components connected to these enclosures are separately installed from above.

2.2.1.6 Control Rods

Separate control rods are used for the functions of shim, startup, and scram. Twelve shim rods are located in the outer part of the core and nineteen startup and scram rods are located in the inner part of the core. These three types of rods differ both in the construction and material of the rods and in the drive mechanisms.

The cross sectional dimension of the rod and the number of components, however, are identical.

The nineteen startup and scram rods are identical in their construction and materials, and are divided into three groups:

1. Seven are operated as primary scram rods.
2. Six are operated on a separate system, and are designated as backup scram rods. They provide redundancy to the scram system.
3. Six are used for startup, and differ in being operated with shim rather than scram drives. Their purpose is for accommodating the reactivity change which accompanies the raising of the reactor to operating temperature. Like the scram rods, they are always withdrawn when the reactor is at power (except during warmup).

Figure 2.8 shows the construction of the control rod and of the surrounding fuel bundle. The dimensions and the number of elements are shown in Figure 2.3 (Figure 2.8 indicates a 7-rod design rather than 19, and a slightly smaller fuel bundle).

Scram and startup rods are always totally withdrawn when the reactor is operating. Boron carbide (B_4C) is utilized in the startup and scram rods because of its very desirable absorption characteristics. The usual problem of helium generation is of little consequence, since the startup and scram rods are required to absorb only a relatively small neutron dose in their lifetime. The B_4C is contained as pellets in groups of 19 tubes (0.600 inch O. D. $\times$ 0.020 wall) to form a poison length of 34 inches. The shim rods are constructed of tantalum. The choice of tantalum shim rods meets the objectives of this study. The choice of B_4C may eventually prove to be more attractive. A discussion of the B_4C rods is presented in the Task III report.

The shim rods are of the same outside dimension as those in the scram rods, except that they are assumed to be solid tantalum enclosed in a bonded cladding of stainless steel which is 0.005 inch thick. Tantalum would be undesirable for scram rods because its effectiveness is somewhat less than that of B_4C and because the rods would have considerably more inertia.

Three guide wheels are provided at the top and at the bottom of the control rods to ride in the corners of the hexagonal guide tubes. This construction assures that the control rod cannot rotate and minimizes the likelihood of binding.

A fitting is provided at the upper end of the control rod to connect the rod to its drive. Disconnection of the control rod from its drive is accomplished by a 90 - degree rotation of the control rod drive shaft. Interference in the fitting prevents a greater shaft rotation. This interference feature provides a check to assure that disconnection has been effected. The procedure for disconnecting is as follows: rotate the shaft 90 degrees with the control rod in the fully inserted position; raise the drive shaft and rotate the shaft an additional 90 degrees. If this added rotation is accomplished without interference, then the control rod is not connected to its drive.

The fuel bundle which contains a control rod is of a construction similar to that of the other fuel bundles, except for the addition of a control rod guide tube which replaces the central 127 fuel pins. The guide tube extends above the top of the fuel bundle by approximately 28 inches. The sides of this extension are provided with three projections which permit the bundle to be handled with the same refueling grapple that handles the other fuel bundles.

The control rod and its fuel bundle are replaced at the same time, since these must be handled as a unit. The discharged control rod may be retrieved in the irradiated fuel storage tank.

2.2.1.7 Control Rod Drives

Two totally different control rod drives are used for the shim and startup rods and for the scram rods. Scram rods utilize a pneumatic drive actuated with reactor cover gas argon which has been cooled and has had sodium vapor removed. Shim and startup rods utilize an electric motor driving a rack and pinion. As with the scram drive, the shim drive operates in reactor cover gas argon in order to eliminate the need for a high integrity sliding seal. Static seals are used in both of these drives to prevent leakage of the reactor cover gas (which contains radioactive xenon and krypton) into the refueling cell. The static seals are installed in pairs with a high pressure buffer of fresh argon trapped between them.

These rod drive designs meet the objectives of this study. Other designs may eventually prove to be more attractive. A discussion of other rod drive designs is presented in the Task III report.

a. Shim and Startup Rod Drive

The design of the shim and startup rod drive is shown in Figure 2.9. The unit is totally sealed, and operates in an atmosphere of cooled reactor cover gas argon. A slight continuous inflow of this gas prevents upward convection of the sodium vapor laden reactor cover gas. Because no shaft seal is required, the rack is permitted to extend below the level of the shield plug. The pinion drive is located at a level immediately above the top of the shield plug. The rack and pinion drive is used rather than a screw drive in order to simplify the connecting and disconnecting. The entire inner mechanism of the drive, including the drive motor and the rack and pinion, is rotated as a unit in order to disconnect the drive from the control rod. The rotation mechanism is located at the top of the drive, and incorporates a positive locking device as well as a visual means of observing the locking position.

FIGURE 2.8 CONTROL ROD AND MODIFIED FUEL BUNDLE (SIMILAR TO GE DRAWING 106D3502)

b. Scram Rod Drive

Figure 2.10 is a schematic diagram of the scram rod drive and of its actuating components. The drive consists of a piston which moves in a stationary cylinder. Movement of the piston results from variations in the pressure of the gas above the piston. Each piston has a long shaft which can be connected to one scram rod. Scram occurs when the quick-opening scram valves expose the top of the pistons to a high-pressure gas reservoir. The space below the piston is vented to the reactor cover gas.

A single header (header No. 1) is used to supply the high-pressure gas to all the scram rods. A single tube at each drive location connects the top of each drive cylinder to the supply header. Part of each tube is permanently cast into the shield plug. The remaining part of each tube is attached to the drive cylinder. The two sections are automatically mated when the drive is installed. A seal ring (seal ring No. 2) is provided at the connection to minimize leakage.

Two features are incorporated in the drive cylinder to decelerate the scram rod at the bottom of its travel. First, a snubber is provided at the bottom of the cylinder to absorb the impact energy. Secondly, the rate at which gas is vented from beneath the piston is suddenly reduced as the piston falls past a vent port. The tube connected to the vent port contains a check valve to prevent inflow of gas to the space above the piston.

Lifting the scram rods requires exposing the tops of the pistons to a low pressure reservoir. For this purpose a second header (header no. 2 on Figure 2.10) is provided. Either of the two headers may be exposed to any scram drive cylinder by opening one of the two valves located at each cylinder (valve no. 1, valve no. 2). The following is the sequence for raising the scram rods from the core:

1. Fully insert all shim rods.

2. Close the scram valves and fill the high pressure tank.
3. Open valve no. 3 and evacuate header no. 1 using vacuum pump no. 1. Valves no. 1 and no. 2 at each drive are closed.
4. Evacuate header no. 2 using vacuum pump no. 2.
5. Open valve no. 2 at one of the scram rod drives. (the valves no. 2 at all of the rod drives are mechanically interlocked so that only one valve no. 2 may be opened) The selected rod rises slowly to the top of the cylinder.
6. A face seal is provided on top of the piston. When the piston contacts the top of the cylinder, the leakage rate of gas past the cylinder is suddenly reduced. A reduction of the pressure in header no. 2 will be noticed as the piston face seal engages.
7. Open valve no. 1 and close valve no. 2. Vacuum pump no. 1 has adequate capacity for the combined gas leakage around the face seals of all of drive pistons. If any one face seal is improperly seated, the increased leakage will increase the pressure in header no. 1 and release all of the drive pistons.
8. Repeat steps 5 through 7 until every scram rod has been raised to its fully withdrawn position.

A third header (header no. 3) is provided to supply treated and recycled reactor cover gas to each of the scram rod drive cylinders. The pressure in this header is high enough to assure continuous slow leakage of argon along the drive shaft and into the reactor cover gas to each of the scram rod drive cylinders. The pressure in this header is high enough to prevent the drive mechanisms. Leakage from the drive cylinder into the refueling cell is

FIGURE 2.10 SCRAM ROD DRIVE

prevented by a seal ring (seal ring no. 1) between the mating surfaces of the drive cylinder and of the shield plug.

The piston is detected in its lowest position by a simple magnetic device built into the shield plug.

One advantage of this scram rod drive arrangement is that the number and complexity of connections which must be made within the refueling cell are minimized.

The drive is connected and disconnected from the rod in a manner similar to that described for the shim rods. The mechanism is not shown on Figure 2.10.

2.2.1.8 Core Composition for Neutronic Calculations

Table 2.1 lists the compositions of the core, radial blanket, and reflector. The individual bundle types are tabulated along with their compositions (areas and percents). In addition to the three basic materials (fuel, steel, and sodium), the control rod hole is listed as part of the control rod fuel bundle. This hole represents the actual size of the inside of the control rod guide tube. The composition of the hole depends upon whether or not it is occupied by a control rod, and its composition is shown in Table 2.2. The control rod is assumed to be made up of 19 rods, each having a diameter of 0.600 inch. Cladding is included in the areas of the rods.

All dimensions are assumed at reactor inlet temperature (810° F), and would be slightly different at room temperature. The nature of the design is such that the radial size of the core is controlled by a surrounding structure which is at reactor inlet temperature. Therefore, compositions at reactor outlet temperature would be the same, except for a correction for the effect of axial expansion. The outlet temperature will actually

cause some radial expansion of the center of the core and compressive deformation of the outer part of the core; thus causing a slight negative reactivity coefficient as the reactor outlet temperature rises.

2.2.2 Reactor Refueling (J. H. Germer)

Refueling of the reactor is carried out once a year and requires that the reactor be shut down for one week. Access to the reactor core is achieved by removing a shield plug from the refueling cell floor. The subsequent operations are carried out in a pre-planned manner with computer control of all fuel accounting, including location of positions in the core. All operations are monitored by direct viewing by the operators. Manual control of the refueling operations is possible in the event of unforeseen difficulties, such as a sticking fuel bundle.

After reactor shutdown and total insertion of all control rods, the entire primary sodium vessel is cooled gradually to 400° F. Partial flow is maintained with the primary circulating pumps to minimize temperature differences.

During this operation the control rod drives are all disconnected by rotating the handles at the top of the drives. As discussed in the section on control rods (2.2.1.6) disconnecting is accomplished by rotating the drives 90 degrees. This is followed by raising the drive and rotating it another 90 degrees to assure that disconnection has taken place. The operations are performed by the telescoping tube overhead manipulator.

Also during cool-down, the cover gas over the reactor is purged. This is accomplished by flowing in fresh argon and exhausting to the radioactive gas handling waste system. To minimize the amount of purge gas required, the sodium level in the reactor vessel is raised by slightly reducing its

TABLE 2.1

LMFBR CORE COMPOSITION

	Fuel Bundle	Control Rod Fuel Bundle	Blanket Bundle	Reflector	Total Core	Total Blanket	Total Reflector
No. of Bundles	234	31	126	72	--	--	--
No. of Pins	331	204	127	--	83778	16002	--
Pin Dia. (Hot) (in.)	0.250	0.250	0.422	--	--	--	--
No. of Wire Spacers	240	168	127	--	61368	16002	--
Wire Diameter (in.)	0.050	0.050	0.050	--	--	--	--
<u>Area (sq. in.)</u>							
Fuel	13.752	8.476	15.327	--	3480.7	1931.2	--
Steel	4.132	4.596	4.990	25.978	1109.4	628.7	1870.4
Sodium	10.214	7.278	7.781	2.120	2615.7	980.4	152.6
Control Rod Hole	--	7.748	--	--	240.2	--	--
Total	28.098	28.098	28.098	28.098	7446.0	3540.3	2023.0
<u>Volume Composition (%)</u>							
Fuel	48.94	30.17	54.55	--	46.74	54.55	--
Steel	14.71	16.36	17.76	92.45	14.90	17.76	92.45
Sodium	36.35	25.90	27.69	7.55	35.13	27.69	7.55
Control Rod Hole	--	27.57	--	--	3.23	--	--
Total	100.00	100.00	100.00	100.00	100.00	100.00	100.00

TABLE 2.2
CONTROL ROD AND GUIDE TUBE COMPOSITION

	Unit Area (sq. in.)	% Of Hole	Area of 31 Units (sq. in.)	% Of Core
RODS (Incl. Cladding)	5.372	69.3	166.6	2.24
SODIUM	2.376	30.7	73.6	0.99
TOTAL (All Sodium with Rods Out)	7.748	100.0	240.2	3.23

cover gas pressure relative to the cover gas pressure over the rest of the primary sodium vessel.

All gas and electrical connections to the shield plug are disconnected by the overhead manipulator. These include multiple connector plugs for thermocouple wiring, and wiring and pipe connections for the control rod drive system. Holddown clamps retaining the shield plug are loosened and rotated aside. The instrumentation probes and the control rod drive extensions are firmly mounted on the lower surface of the shield plug (see Section 2.2.3.2), and are not disturbed as the plug is raised. The proper location of the probes and extensions when the plug is in position is assured by the close control of dimensional tolerances during fabrication.

The shield plug is now raised at a very slow rate for the first 3 feet as a final precaution against the extremely remote possibility that some control rods are still connected to their drives. The shield plug is then lifted its entire distance at a more rapid rate. Lifting motion is achieved by two ball-nuts on vertical lead screws driven in synchronism by a motor outside of the refueling cell.

Figure 2.11 shows schematically the entire refueling sequence, beginning with the bringing in of new fuel to the refueling cell while the reactor is still operating, and ending

with removal of old fuel from the refueling cell after the reactor operation is resumed. The steps in the refueling sequence are tabulated below:

Summary of Steps Shown in Refueling Schematic - (Figure 2.11)

1. Bring new fuel into the refueling cell in transfer cups during reactor operation.
2. Raise the shield plug after the reactor has been shutdown and cooled.
3. Transfer new fuel into temporary storage positions around the core (23 cups containing fuel bundles plus 1 empty cup).
4. Transfer one used fuel assembly from the core to the empty cup in the temporary storage position (using fuel shuffling machine).
5. Transfer one new fuel assembly from the temporary storage position to the empty position in the core (using fuel shuffling machine).

Repeat steps 4 and 5 until 23 fuel bundles have been replaced (note that only one bundle is removed from the core at any one time).
6. Transfer the used fuel to the decay tank

FIGURE 2.11 REFUELING SEQUENCE - SCHEMATIC DIAGRAM

in the transfer cups (23 fuel bundles). Repeat the sequence from step 3 to step 6 eight times to complete the refueling operation.

7. Replace the shield plug, and resume reactor operation.
8. Transfer the used fuel for shipment from the decay tank during reactor operation.

The following points should be especially noted:

- (1) All transfer operations of used fuel in the refueling cell are done in transfer cups filled with sodium, and lifting is accomplished with a grapple which includes a secondary natural convection heat transfer loop (steps 3, 6, 8).
- (2) New fuel is handled, within the refueling cell, in transfer cups which do not contain sodium (steps 1 and 3).
- (3) Transfer of fuel between the core and the temporary storage positions is done by the fuel shuffling machine. The fuel is totally immersed in the reactor sodium during these transfers (steps 4 and 5). This machine is retracted into the end of the refueling cell when other transfer operations are done (steps 3 and 6).
- (4) Transfer of new fuel into the refueling cell and of used fuel out (steps 1, 8) are normally done with the reactor in operation, and therefore the time required does not affect the reactor down time.

The refueling time estimate is shown in Table 2.3. A total shutdown time of one week is predicted for refueling the reactor. For each step the time after reactor shutdown at which the operation is started and is finished is given in the table. Note that some of the operations, such as cooling the

reactor and disconnecting control rod drives, are done simultaneously. Several assumptions are made:

1. All operations are performed without major equipment breakdown.
2. Repairs on the thermocouples or control rod drives are not made in place. The need for repairs is anticipated and a replacement unit is available at shutdown. The estimate includes enough time to replace eight control rod drives.
3. The refueling operation is completely preplanned, and all record keeping and machine location operations are computer controlled rather than manual. Operator control is limited to monitoring, and possibly fine control of location adjustments.
4. Refueling is done around the clock.
5. No operations are done with the reactor shutdown which could be done during reactor operation. This includes bringing new fuel bundles into the cell and removing old fuel bundles from the cell.
6. No personnel are required to enter the refueling cell during the refueling period.
7. Fuel bundles are each handled and stored in the refueling cell within separate fuel transfer thimbles. They are not removed from these thimbles except within the reactor vessel.
8. The fuel and radial blanket bundles are refueled on the following schedule:

54 fuel bundles = 18 units
(3 year cycle) per year

211 fuel bundles = 106 units
(2 year cycle) per year

TABLE 2.3

REFUELING TIME ESTIMATE

	Time From Shutdown (hrs)	
	Start	Finish
Cool reactor and primary vessel	0	9
Purge Argon cover gas	7	9
Disconnect control rod drives (31 units at 13.5 min each)	2	9
Disconnect wires and tubes from shield plug	9	10
Replace control rod drives and outlet thermocouples (assume 8 control rod drives at 1 hour each)	10	18
Unfasten shield plug	18	19
Raise shield plug	19	22
Transfer 23 thimbles with bundles and one empty thimble to core vessel (10 min each)	22	26
Shuffle 23 fuel bundles between thimbles and core (46 operations at 10.5 min each)	26	34
Remove 23 used bundles in cups to decay tank (23 operations at 10.5 min each) (at last time, remove empty thimble also)	34	38
The previous three steps performed seven times (7 operations at 16 hours each)	22	134
Prepare for shield plug replacement, e.g. move fuel shuffling machine to other side of cell	134	135 (5.6 days)
Lower shield plug	135	138
Clamp shield plug, check seals, etc.	138	142
Reconnect wires and tubes to shield plug and check them	142	144
Reconnect control rod drives (31 units at 11.5 minutes)	144	150
Withdraw shim rods until criticality is reached	152	156
Raise temperature and establish power and steam flow	156	168
Total		168 hrs. (7 days)

60 blanket bundles (3 year cycle) = 20 units per year

63 blanket bundles (4 year cycle) = 16 units per year

Total bundles re-fueled = 160 units per year

No allowance is made for replacement of reflector bundles. These may be replaced very infrequently, such as once in ten years.

2.2.2.1 Fuel Shuffling Machine

The fuel shuffling machine transfers fuel bundles under sodium between positions in the reactor core and the transfer thimbles. The transfer thimbles are located in the 24 temporary locations surrounding the reactor. In addition to fuel bundles, the machine can handle both radial blanket and reflector guide bundles. At no time does the fuel shuffling machine raise fuel above the level of the sodium. The machine can be retracted when not in use to a position above the floor of the refueling cell and totally out of the way of the fuel transfer machine.

The fuel shuffling machine (see Figure 2.12) operates on an L-shaped carriage with wheels which roll on rails at two sides of the refueling cell. A cross carriage travels on one horizontal box beam of the L-shaped carriage. The L-shaped carriage is propelled by a horizontal lead screw which extends over its full length of travel. The cross carriage is propelled by another horizontal lead screw which is perpendicular to the main lead screw. The lead screw of the cross carriage is powered through a gear box which slides along a splined shaft. The splined shaft is parallel to the main lead screw and is of equivalent length. The fuel shuffling

machine may be moved to any point in a horizontal plane above the reactor by rotating the main lead screw and the splined shaft. Both of the major shafts penetrate the end of the refueling cell through seals and can be rotated by hand if necessary. The other motions of the fuel shuffling machine are electrically driven.

The cross carriage supports a vertical hexagonal tube (see Figure 2.13) which is slightly larger in size than a fuel bundle. A vertical motion is imparted to this hexagonal tube by means of a motor-driven rack and pinion with a stroke sufficient to move the lower end of the hexagonal tube from the top of the fuel bundles to a retracted position above the floor of the refueling cell. Attached to the upper end of the hexagonal tube is a motor-driven pinion which engages a rack for lifting a grapple which is located within the hexagonal lifting tube. The grapple moves a distance, relative to the tube, which is slightly greater than the length of the fuel bundles. This motor-driven grapple is provided with a separate motion which rotates the grapple unit over a 60-degree stroke to latch onto the fuel bundle handles. This 60-degree motion is possible only with the grapple in its lowest position because of an intentional interference of the grapple within the hexagonal tube. Cutouts are provided in the hexagonal tube to allow rotation only in the lowest position (see Figure 2.14).

The operation of the fuel shuffling machine consists of the following steps:

1. Lower the hexagonal tube until it rests on the fuel bundles which surround the one to be placed,
2. Rotate the grapple to engage the fuel bundle handle,
3. Raise the fuel bundle until it is entirely within the hexagonal tube,

FIGURE 2.12 FUEL SHUFFLING MACHINE - ARRANGEMENT

FIGURE 2.13 FUEL SHUFFLING MACHINE - SECTION

FIGURE 2.14 FUEL SHUFFLING MACHINE GRAPPLE OPERATION

4. Raise the hexagonal tube to clear the adjacent fuel bundle handles,
5. Transfer the unit horizontally until it is directly over an empty fuel transfer thimble,
6. Lower the hexagonal tube to rest on the top of the fuel transfer thimble,
7. Lower the grapple until the fuel bundle is seated at the bottom of the fuel transfer thimble,
8. Rotate the grapple to disengage, and, finally, remove the fuel shuffling machine to perform its next operation.

The operation is similar for moving fuel from the fuel transfer thimble to a position in the core. In making these transfers, the horizontal motions must be programmed to avoid interference with projecting control rod guide tube extensions. When a fuel bundle containing a control rod is to be picked up, the hexagonal tube must be lifted higher to fit over the control rod guide tube extensions. However, this is required only when a fuel bundle is not contained within the hexagonal tube. When transferring this special fuel bundle, only the normal lift is required. The grapple design provides a recess in the center to permit the picking up of either type of handle.

It should be noted that this method of raising fuel bundles precludes the possibility of raising an adjacent fuel bundle accidentally. Alignment of the fuel shuffling machine over the desired fuel bundle does not require extreme precision, since the hexagonal tube automatically centers itself on the handles of three of the adjacent fuel bundles. The hexagonal tube can not be seated except in a position over a fuel bundle.

Accidental insertion of a regular fuel bundle without a control rod in a space nor-

mally occupied by one with a control rod is prevented by using a slightly larger lower end fitting on fuel bundles with control rods. Also, the grapple may not be disengaged in other than its fully extended position.

It should be noted that the hexagonal tube of the fuel shuffling machine is not capable of rotation. Therefore, its faces are always parallel to those of the fuel bundles in the core. In order to assure proper angular orientation of the three-bladed handles of the fuel bundles, the bundles are always installed in the fuel transfer thimbles in the same orientation with respect to the off-center location of the guide pins of the thimble. These guide pins assure that the thimbles can only be placed in positions which result in a correct orientation for the fuel bundle. Since the 24 storage positions are in two angular positions in respect to the three-sided symmetry of the fuel bundle handles, the thimble guides on alternate sides of the core are located toward the front and toward the back, respectively (see Figure 2.6).

2.2.2.2 Fuel Transfer Machine

The fuel transfer machine transfers fuel bundles, along with the fuel bundle transfer thimbles, between the temporary storage positions around the core and either the fuel decay tank or the new-fuel storage tank. When the transfer is from the reactor vessel, the transfer thimbles remain filled with sodium to assure adequate heat transfer from the highly radioactive fuel bundle. The high heat generation of the used fuel bundle (as much as 50 kilowatts each) requires additional heat removal capacity in the event of a delay during the transfer. This heat transfer is provided by a heat exchanger built as an integral part of the fuel transfer machine grapple (see Figure 2.15). The grapple, whose outer surface is covered with fins, is filled with NaK and sealed. It receives heat through two dip tubes which extend into the two sides of the fuel transfer thimble. A

natural convection circuit of NaK within this grapple transfers heat to the fins. The fins in turn transfer heat to the refueling cell by natural convection to the argon and by radiation. As a safety precaution, the NaK system is contained in two sections such that either side could drain without draining the other in the event of a leak.

In normal fuel transfer, the heat capacity of the unit limits the temperature rise to approximately 100° F over the temperature of the reactor vessel sodium. If the mechanism for transferring the fuel bundle fails to operate during the transfer, the temperature will slowly rise until the fuel reaches an equilibrium temperature of approximately 1100° F after an hour. Lower temperatures would occur if a circulating fan is

utilized to supplement the very poor natural convection properties of argon. Table 2.4 summarizes the heat transfer characteristics of the fuel transfer grapple.

The heat transfer fuel grapple is attached to a telescoping tube crane manipulator, which permits the grapple to extend to the level of the 24 fuel transfer thimbles surrounding the core, raise these thimbles out of the reactor vessel, and transfer them to the fuel decay tank. Control of angular position about the vertical axis is required in order to fit both the storage racks in the fuel decay tank and the positions around the core.

The heat transfer fuel grapple fits within a cylindrical guide tube over each of the 24 temporary storage positions around the

TABLE 2.4
HEAT TRANSFER CHARACTERISTICS
OF THE FUEL TRANSFER GRAPPLE

NaK natural circulation tubing i. d.	0.78 in.
NaK natural circulation flow rate	570 lb/h
Average heat transfer coefficient at fin surface	0.7 Btu/h ft ² °F
Number of fins	68
Fin length	17 ft.
Fin width	3 in.
Total heat transfer area*	297 ft ²
Fuel bundle maximum decay power	50 l /kW

* Does not include area of cup holding fuel which contributes about 15% of the heat transfer

FIGURE 2.15 FUEL TRANSFER GRAPPLE AND FUEL TRANSFER THIMBLE (SIMILAR TO GE DRAWING 277E201)

core. These tubes extend from the level of the top of the core to slightly above the sodium level, and are open on one side to permit horizontal transfer of fuel by the fuel shuffling machine. Where these tubes are located near the sodium outlet pipes, they are further perforated to minimize flow interference.

At the sides of the reactor, the cylindrical guide tubes extend downward, but are modified to an oval shape to accommodate the fuel transfer thimbles. Two tracks extend to the bottom of the oval holes to guide the fuel transfer thimbles and the fuel transfer grapple. Each fuel transfer thimble has three pins, two of which slide in one track, and a single one at the bottom which slides in the other track. This three-pin arrangement provides adequate guidance to assure proper functioning of the grapple, and offers the least opportunity for binding. As noted before, the tracks as well as the guide pins are located asymmetrically in order to permit installation only in the proper orientation. Units on alternate sides of the reactor are located with opposite orientation so that the three-bladed fuel bundle handles will be properly oriented with respect to the core (see Figure 2.6).

Storage racks in the fuel decay tank are provided with similar guide tracks to assure accurate alignment of the grapple during insertion and removal of fuel transfer thimbles. The storage racks provide positions for 174 fuel bundles (or radial blanket bundles) arranged in five concentric circles (Figure 2.16). The new fuel storage location has 174 positions in the same arrangement.

2.2.3 Reactor Performance

The nuclear and the thermal and hydraulic characteristics of the reactor core were carefully selected to minimize the total cost of energy from the LMFBR power plant. A detailed description of the procedures and assumptions which led to the selection of the core design will be presented in the Task III report.

The isotopic distribution of the plutonium, which is used to enrich the core, is typical of the plutonium which will be discharged from light-water moderated and cooled thermal reactors. It is expected that plutonium from the thermal reactors will be the dominant fuel for fast reactors which are constructed for operation in the 1980's. The effect on performance and on economics of using plutonium with a different isotopic distribution will be reported in the Task III report.

The neutron cross section data which were used in selecting the core design are typical of those currently being used in the United States except for modifications to the capture-to-fission ratio of Pu-239 and the capture cross section of U-238. The modifications reflect recent measurements which imply higher values for the capture-to-fission ratio. A detailed description of the modifications is presented in the Task I Report. The effect on performance and on economics of using the unmodified cross-section data will be reported in the Task III report.

2.2.3.1 Nuclear Performance (C. L. Cowan)

The nuclear design of the LMFBR core evolved from the advanced core model described in the Task I report of the LMFBR Project.⁽¹⁾ The principal differences between the Task I and Task II cores ensued from a detailed evaluation of the inter-relationship between fuel density, fuel exposure, and fuel-structure ratio. A description of the screening calculations which were used to guide the selection of the Task II design are presented in the Task III Final Report. New basic nuclear data which indicated a likelihood of larger values for the Pu-239 capture-to-fission ratio were also taken into account during the Task II effort.

The LMFBR core is characterized by a high fuel volume fraction and low structure content which is consistent with the assumed

state-of-art in 1980. The core was designed with the emphasis on economic performance and depends upon instrumentation and control systems to assure complete safety. Because of this approach to safety, the sodium void reactivity effects are considerably more positive than the void reactivity effects described in earlier design studies. ^(2,3) Similarly, a minimum absolute value of the Doppler coefficient was not required to satisfy design safety criteria.

a. Core and Blanket Fuel Management

1. Core and Blanket Arrangement. The core and blanket arrangement for the LMFBR reference design is shown in Figure 2.17. The core has a height-to-diameter ratio of about 1 to 3. The axial blanket thickness is based upon the trade-off between fuel cycle economics and total pressure drop. Similarly, the radial blanket thickness, corresponding to two rows of fuel subassemblies, is based upon a trade-off between fuel cycle economics and overall plant size (e.g., the shield plug).

The axial blanket assembly is an integral extension of the core assembly. However, the fuel densities in the core and blanket are different. The oxide fuel density used in the core was 8.93 gm/cc (85% of the theoretical fuel density at an average fuel temperature of about 2670°F) whereas the fuel density in the axial and radial blankets was 10.0 gm/cc. Volume fractions for all core and blanket components are listed in Figure 2.17.

2. Fuel Management for the Equilibrium Fuel Cycle. Radial power flattening is achieved by varying the fissile plutonium content in two radial core zones of equal volume. Core Zone 2 (Figure 2.17) was further divided into two sub-zones of different fuel residence times to achieve a uniform discharge exposure. Fuel management during the equilibrium fuel cycle is consistent with the following assumptions:

1. The supplied fuel for the core is discharged and reprocessed plutonium from a water reactor system,
2. The supplied fuel for the blankets is depleted uranium (U-238 with 0.3 atom percent of U-235),
3. The load factor is 0.8, and
4. All core fuel is irradiated at a fixed location to an average discharge exposure, by zone and sub-zone, of approximately 100,000 MWd/Te.

The first assumption is based upon a forecast of the available plutonium in the 1980's which has indicated that the predominate plutonium supply for this period will come from water reactor systems. The isotopic composition of the water reactor plutonium is sensitive to the discharge exposure and is also influenced by the reactor spectrum and shielding effects (e.g., near the large Pu-240 resonance at approximately 1 eV). Plutonium isotopic fractions for a BWR discharge exposure of about 20,000 MWD/Te are given as follows:

Pu-239	0.589
Pu-240	0.257
Pu-241	0.122
Pu-242	0.032

A nominal twelve month refueling interval was specified for the LMFBR design and for the exposure and load factor indicated above approximately 45.8% of the core is refueled at the end of each fuel cycle. The reloading scheme for the equilibrium fuel cycle is shown in Table 2.5.

Based upon the reloading scheme in Table 2.5, the average discharge exposure of the core fuel was computed to be 100,167 MWd/Te.

3. Fuel Isotopic Compositions. The average fuel isotopic compositions for the various regions of the core and blankets as they are

FIGURE 2.16 REFUELING CELL ARRANGEMENT
(SIMILAR TO GE DRAWING 27E189)

COMPOSITION (VOLUME PERCENT)

	CORE AND AXIAL BLANKET	RADIAL BLANKET
FUEL	46.74	54.55
SODIUM	35.13	17.76
STRUCTURE	14.90	27.69
CONTROL	3.23	0.00
	<u>100.0</u>	<u>100.0</u>

FIGURE 2.17 LMFBR CORE AND BLANKET ARRANGEMENT

supplied and discharged are presented in Tables 2.6 and 2.7. Tables 2.8 through 2.10 list the average regional fuel isotopic compositions at the beginning, middle and end of the 12-month burnup interval, respectively, for these core and blanket regions. The symbolism used in these tables is consistent with Figure 2.17 and is described as follows:

CZ1 Core Zone 1

CZ2 Core Zone 2 is divided into two sub-zones of equal volume. The fuel bundles positioned along the inner radius of Core Zone 2 are labelled CZ2

CZ3 The fuel bundles positioned along the outer radius of Core Zone 2 are labelled CZ3

AB1 Axial blankets above and below Region CZ1

AB2 Axial blankets above and below Region CZ2

AB3 Axial blankets above and below Region CZ3

RB { RB1 The radial blankets consist of two rows of fuel subassemblies. The fuel bundles adjacent to the core are labelled RB1.
RB2 The fuel bundles positioned along the outer radius of the radial blankets are labelled RB2.

CRB { CB1 Blanket regions positioned above and below Radial Blanket Region RB1
CB2 Blanket regions positioned above and below Radial Blanket Region RB2

TABLE 2.5

RELOADING SCHEME FOR AN EQUILIBRIUM FUEL CYCLE

	Fraction of Fuel Replaced	Average Residence Time (yr)	Average Discharge Exposure (MWd/Te)	Fissile Inventory Discharged (kg of Pu-239, Pu-241 and U-235)	Fissile Inventory Supplied (kg of Pu-239 Pu-241 and U-235)
CZ1	1/2	2.0	101,265	357.0	358.6
CZ2 ^(a)	1/2	2.0	98,614	206.1	250.3
CZ3 ^(a)	1/3	3.0	99,523	137.1	166.9
AB1 ^(b)	1/2	2.0	9,146	143.6	11.1
AB2 ^(b)	1/2	2.0	5,784	56.6	5.5
AB3 ^(b)	1/3	3.0	6,342	40.0	3.7
(RB1 + CB1) ^(c)	1/3	3.0	8,664	94.5	7.8
(RB2 + CB2) ^(c)	1/4	4.0	4,931	64.4	6.4
TOTAL				1,099.3	810.3

NOTES:

- Core Zone 2 is divided into two subzones of approximately equal volumes. The fuel bundles along the inner radius are labeled CZ2. The fuel bundles along outer radius are labeled CZ3.
- The axial blankets use the same numbering system as the corresponding core regions.
- The fuel bundles along the inner radius of the radial blanket are labeled (RB1 + CB1). The fuel bundles along the outer radius of the radial blanket are labeled (RB2 + CB2).

TABLE 2.6

AVERAGE FUEL ISOTOPIC COMPOSITIONS SUPPLIED* (ATOM PERCENT)

	<u>Pu-239</u>	<u>Pu-240</u>	<u>Pu-241</u>	<u>Pu-242</u>	<u>F. P.</u>	<u>U-235</u>	<u>U-238</u>
CZ1	8.84	3.86	1.83	0.48	0.0	0.0	84.99
CZ2	12.35	5.39	2.56	0.67	0.0	0.0	79.03
CZ3	12.34	5.39	2.56	0.67	0.0	0.0	79.04
AB1	0.0	0.0	0.0	0.0	0.0	0.30	99.7
AB2	0.0	0.0	0.0	0.0	0.0	0.30	99.7
AB3	0.0	0.0	0.0	0.0	0.0	0.30	99.7
RB1	0.0	0.0	0.0	0.0	0.0	0.30	99.7
RB2	0.0	0.0	0.0	0.0	0.0	0.30	99.7
CB1	0.0	0.0	0.0	0.0	0.0	0.30	99.7
CB2	0.0	0.0	0.0	0.0	0.0	0.30	99.7

*Based upon the fuel loading schedule shown in Table 2.5

TABLE 2.7

AVERAGE DISCHARGE FUEL ISOTOPIC COMPOSITIONS* (ATOM PERCENT)

	<u>Pu-239</u>	<u>Pu-240</u>	<u>Pu-241</u>	<u>Pu-242</u>	<u>F. P.</u>	<u>U-235</u>	<u>U-238</u>
CZ1	9.75	4.32	0.89	0.51	10.53	0.0	74.00
CZ2	10.93	5.54	1.35	0.71	10.25	0.0	71.22
CZ3	10.97	5.60	1.28	0.71	10.35	0.0	71.09
AB1	4.14	0.37	0.01	0.00	0.96	0.20	94.32
AB2	2.79	0.11	0.00	0.00	0.60	0.21	96.29
AB3	2.98	0.14	0.00	0.00	0.66	0.21	96.01
RB1	4.56	0.26	0.01	0.00	1.49	0.17	93.51
RB2	3.63	0.21	0.01	0.00	0.80	0.19	95.16
CB1	2.14	0.08	0.00	0.00	0.30	0.23	97.25
CB2	1.83	0.08	0.00	0.00	0.22	0.23	97.64

*Based upon the fuel loading schedule shown in Table 2.5

TABLE 2.8

AVERAGE FUEL ISOTOPIC COMPOSITIONS AT THE
BEGINNING OF THE EQUILIBRIUM FUEL CYCLE* (ATOM PERCENT)

	<u>Pu-239</u>	<u>Pu-240</u>	<u>Pu-241</u>	<u>Pu-242</u>	<u>F. P.</u>	<u>U-235</u>	<u>U-238</u>
CZ1	9.16	3.97	1.54	0.49	2.63	0.0	82.21
CZ2	11.98	5.43	2.20	0.69	2.56	0.0	77.14
CZ3	11.88	5.47	2.05	0.69	3.45	0.0	76.46
AB1	0.97	0.03	0.0	0.0	0.24	0.27	98.49
AB2	0.61	0.02	0.0	0.0	0.15	0.28	98.94
AB3	1.06	0.03	0.0	0.0	0.22	0.27	98.42
RB1	1.66	0.06	0.0	0.0	0.50	0.25	97.53
RB2	1.46	0.06	0.0	0.0	0.30	0.26	97.92
CB1	0.74	0.02	0.0	0.0	0.10	0.27	98.87
CB2	0.71	0.02	0.0	0.0	0.08	0.27	98.92

*Based upon the fuel loading schedule shown in Table 2.5

TABLE 2.9

AVERAGE FUEL ISOTOPIC COMPOSITIONS AT MID-BURNUP OF THE
EQUILIBRIUM FUEL CYCLE* (ATOM PERCENT)

	<u>Pu-239</u>	<u>Pu-240</u>	<u>Pu-241</u>	<u>Pu-242</u>	<u>F. P.</u>	<u>U-235</u>	<u>U-238</u>
CZ1	9.48	4.09	1.25	0.51	5.26	0.0	79.41
CZ2	11.62	5.48	1.85	0.70	5.13	0.0	75.22
CZ3	11.64	5.52	1.80	0.70	5.17	0.0	75.17
AB1	1.95	0.07	0.00	0.00	0.48	0.24	97.26
AB2	1.48	0.04	0.00	0.00	0.30	0.26	97.92
AB3	1.58	0.05	0.00	0.00	0.34	0.25	97.78
RB1	2.48	0.09	0.00	0.00	0.74	0.23	96.46
RB2	1.95	0.07	0.00	0.00	0.40	0.24	97.34
CB1	1.10	0.03	0.00	0.00	0.15	0.26	98.46
CB2	0.96	0.03	0.00	0.00	0.10	0.26	98.65

*Based upon the fuel loading schedule shown in Table 2.5

TABLE 2.10

AVERAGE FUEL ISOTOPIC COMPOSITIONS AT THE
END OF THE EQUILIBRIUM FUEL CYCLE* (ATOM PERCENT)

	<u>Pu-239</u>	<u>Pu-240</u>	<u>Pu-241</u>	<u>Pu-242</u>	<u>F. P.</u>	<u>U-235</u>	<u>U-238</u>
CZ1	9.62	4.21	1.07	0.51	7.90	0.0	76.69
CZ2	11.27	5.51	1.60	0.70	7.69	0.0	73.23
CZ3	11.42	5.54	1.63	0.70	6.90	0.0	73.81
AB1	2.79	0.13	0.00	0.00	0.72	0.22	96.14
AB2	2.13	0.08	0.00	0.00	0.46	0.20	97.13
AB3	2.05	0.08	0.00	0.00	0.44	0.24	97.19
RB1	3.17	0.15	0.00	0.00	1.00	0.21	95.47
RB2	2.37	0.11	0.00	0.00	0.51	0.23	96.78
CB1	1.46	0.04	0.00	0.00	0.20	0.25	98.05
CB2	1.17	0.04	0.00	0.00	0.14	0.26	98.39

*Based upon the fuel loading schedule shown in Table 2.5

The isotopic compositions in Table 2.6 through 2.10 correspond to the fuel loading scheme described in Table 2.5.

4. Neutron Balance Data. Table 2.11 gives a tabulation of the neutron balance data for the LMFBR system. The neutron balance data are based upon the mid-burnup condition of the equilibrium fuel cycle. The region indicators are the same as those used in the previous section except that the two radial blanket regions, RB1 and RB2, have been combined into region RB, and the two corner blanket regions, CB1 and CB2, have been combined into CRB. The capture events in the axial and radial steel reflector have also been included in Table 2.11.

5. Average Fissile Inventory, Breeding Ratio, and Doubling Time. The fissile inventories for the core and blankets, averaged over the equilibrium fuel cycle, are listed as follows:

Core	1624 kg
Axial Blankets	299 kg
Radial Blankets	312 kg
Total	2235 kg

The core inventory corresponds to an average specific power of 1304 kW/kg of fissile material.

TABLE 2.11
NEUTRON BALANCE DATA FOR THE LMFBR DESIGN

EVENTS	CZ1	CZ2	CZ3	AB1	AB2	AB3	RB	CRB	REF	TOTAL
CAPTURES										
F. P.	1.397	0.506	0.365	0.078	0.019	0.016	0.063	0.009	0	2.453
B.	0 0	0	0	0.225	0.080	0.059	0	0	0	0.364
C.	0	0	0	0	0	0	0	0	0	0
Ox	0.107	0.048	0.031	0.019	0.007	0.005	0.015	0.002	0	0.234
Na	0.110	0.040	0.028	0.046	0.017	0.012	0.022	0.010	0.040	0.325
Cr	0.191	0.074	0.051	0.064	0.023	0.017	0.057	0.022	0.268	0.767
Fe	0.891	0.335	0.232	0.337	0.122	0.088	0.288	0.122	1.335	3.750
Ni	0.219	0.090	0.060	0.059	0.021	0.015	0.053	0.017	0.227	0.761
Ta	0	0.516	0.359	0.000	0.175	0.125	0	0	0	1.175
U-235	0	0	0	0.053	0.021	0.015	0.038	0.025	0	0.152
U-236	0	0	0	0	0	0	0.002	0	0	0.002
U-238	15.320	5.387	3.728	7.905	2.896	2.091	5.769	2.597	0	45.702
Pu-239	2.819	1.226	0.867	0.378	0.108	0.088	0.289	0.092	0	5.867
Pu-240	1.128	0.554	0.390	0.014	0.003	0.002	0.011	0.003	0	2.105
Pu-241	0.332	0.180	0.122	0	0	0	0	0	0	0.634
Pu-242	0.186	0.096	0.067	0	0	0	0	0	0	0.349
TOTAL	22.709	9.052	6.300	9.178	3.492	2.533	6.607	2.899	1.870	64.640
FISSIONS										
U-235	0	0	0	0.132	0.051	0.037	0.096	0.053	0	0.369
U-238	2.343	0.984	0.634	0.515	0.189	0.128	0.425	0.061	0	5.279
Pu-239	11.022	5.325	3.611	1.056	0.304	0.236	0.828	0.191	0	22.573
Pu-240	0.953	0.555	0.366	0.001	0.001	0.000	0.004	0	0	1.880
Pu-241	2.051	1.167	0.778	0	0	0	0.001	0	0	3.995
Pu-242	0.100	0.058	0.038	0	0	0	0	0	0	0.196
TOTAL	16.379	8.089	5.425	1.704	0.545	0.401	1.354	0.305	0	34.292
Radial Leakage							0.371	0.081		0.452
Axial Leakage				0.394	0.094	0.108		0.057		0.616
TOTAL										100.000

Contributions to the total breeding ratio for the two core zones and the axial and radial blankets are listed in Table 2.12. The breeding ratio is defined, according to contract specifications⁽⁶⁾, as the ratio of the rate of production of fissile atoms, by capture in fertile atoms, to the rate of destruction of fissile atoms, by capture and fission. The ratio is to be taken over the entire reactor and at the mid-point of a burnup cycle, for the equilibrium core and blanket fuel cycle. The total breeding ratio was computed in this manner to be 1.423.

Core conversion ratios were also determined from the fertile capture rates and fis-

sile absorption rates. Fissile conversion ratios for Core Zones 1 and 2 are 1.01 and 0.76 respectively.

Mid-cycle capture and fission rates for all core and blanket materials are listed on Table 2.11.

The doubling time of the fissile fuel is 7.75 years consistent with an out-of-pile reprocessing time of nine months. This figure is based upon an exponential doubling time expression which is described in Reference 7.

TABLE 2.12

**REGIONAL BREEDING RATIO CONTRIBUTIONS AND
CORE CONVERSION RATIOS^a**

	<u>Contribution to Total Breeding Ratio Above Unity^b</u>	<u>Core Conversion Ratio^c</u>
Core Zone 1	0.007	1.01
Core Zone 2	-0.096	0.76
Axial Blankets above and below Core Zone 1	0.188	
Axial Blankets above and below Core Zone 2	0.123	
Radial Blankets	0.201	
Total	0.423	

Total Breeding Ratio = 1.423

- a. The fissile conversion ratios presented in this table are based upon mid-cycle reaction rates and thus, are slightly different than the fissile conversion ratios implied by a mass balance using the supplied and discharged fissile inventories from Table 2.5.
- b. Defined as $(C_i - A_i) / \sum_i A_i$ where C_i represents the mid-cycle capture rate for U-238 + Pu-240 in Region i and A_i is the mid-cycle absorption rate for Pu-239 + Pu-241 + U-235 in Region i.
- c. Defined as C_i / A_i using the same nomenclature as b above.

b. Power Distribution and Peaking Factors

1. Spectral Flux and Power Distribution.

Figure 2. 18 is a histogram of the relative power per unit lethargy in the core as a function of lethargy. The total power is normalized to 100 percent and is consistent with the mid-burnup condition of an equilibrium fuel cycle. The power spectrum at different times over the 12-month burnup interval is not significantly different from that shown in Figure 2. 18. Energy group numbers are listed on the histogram.

Detailed features of the power spectrum in the core can be further described by spectral indices as follows:

Fissions Below 5.5 keV (%)	8.67
Mean Fission Energy (keV)	199.8

The percent of the fissions due to neutrons whose energies are less than 5.5 keV is given by the first index, and the value marking the energy at which equal numbers of higher and lower energy neutrons cause fissions is given by the last index.

Figure 2. 19 is a histogram of the relative neutron flux per unit lethargy for Core Zones 1 and 2. The fluxes for each zone were again normalized to 100 percent. The spectrum is appreciably hardened in going from Zone 1 to Zone 2 because of the increased fissile enrichment in the latter zone.

The effect of the sodium resonance at ~ 2.85 keV on the flux spectrum (and power spectrum) is shown at a lethargy of eight in the histogram. This resonance produces a dip in the spectra for energy group 10, and a rise for group 11 which is just below the resonance energy.

Figure 2. 20 is a histogram of the relative adjoint fluxes at the center of the core. The adjoint flux, which is a measure of neutron importance at a given energy, drops off sharply

below the U-238 fission threshold and continues to decrease gradually to about 3 keV, reflecting the increasing alpha values (σ_c/σ_f) with decreasing energy and also the increasing absorption rate in fertile and parasitic materials relative to the fission rate.

2. Power and Neutron Flux Peaking Factors.

The total power in the LMFBR design and the change in the power distribution during an equilibrium fuel cycle are given in Table 2. 13. These results show that approximately 8% of the core power shifts to the blankets during the 12-month burnup period with most of the power shift going to the axial blankets.

Table 2. 13 also gives the mid-cycle radial and axial peak-to-average power ratios for the design core. Power flattening in the radial direction is achieved by varying the fissile enrichment in two radial core zones of equal volume. The radial power distribution in the core and radial blankets at mid-burnup of the equilibrium cycle is illustrated in Figure 2. 21.

The radial peak-to-average power ratio in the core can be assumed to remain constant during the burnup cycle in spite of the different fissile conversion ratios in Core Zones 1 and 2 (Table 2. 12) and also in spite of the power shift to the radial blankets. This is because the burnup control rods are selectively positioned and withdrawn during the fuel cycle such that the relative radial power distribution in the core is not changed.

The axial peak-to-average core power ratio is reduced by about 4% during the burnup cycle corresponding to beginning and end of cycle axial peaking factors of 1.22 and 1.17 respectively.

The average and peak values of the total neutron flux and the neutron flux above 825 keV are given in Table 2. 14. The flux above 825 keV is an important indicator of possible neutron irradiation damage and represents about 13% of the total flux. Based upon a fuel

FIGURE 2.18 POWER SPECTRUM AT MID-BURNUP OF AN EQUILIBRIUM CYCLE

FIGURE 2 19 FLIIX SPECTRUM IN CORE ZONES 1 AND 2 AT MID-BURNUP OF AN EQUILIBRIUM FUEL CYCLE

FIGURE 2.20 ADJOINT FLUX SPECTRUM AT MID-BURNUP OF AN EQUILIBRIUM FUEL CYCLE

FIGURE 2.21 RADIAL POWER DISTRIBUTION

TABLE 2.13

REGIONAL DISTRIBUTION OF TOTAL POWER AND
POWER PEAKING FACTORS

	<u>Power at Mid-Burnup of an</u> <u>Equilibrium Cycle (MW_t)</u>		
Core Zone 1	1167		
Core Zone 2	950		
Axial Blankets Above and Below Core Zone 1	118		
Axial Blankets Above and Below Core Zone 2	65		
Radial Blankets	<u>116</u>		
Total	2416		

	<u>Power Distribution During the</u> <u>Equilibrium Fuel Cycle (% of total power)</u>		
	<u>Beginning</u> <u>of Cycle</u>	<u>Mid-</u> <u>Cycle</u>	<u>End of</u> <u>Cycle</u>
Core	90.9	87.6	84.4
Axial Blankets	5.1	7.6	10.0
Radial Blankets	4.0	4.8	5.6
Mid-Cycle Radial Peak-to-Average Power Ratio			1.22
Mid-Cycle Axial Peak-to-Average Power Ratio			1.20

TABLE 2.14

AVERAGE AND PEAK NEUTRON FLUXES

	<u>Neutron Flux in the Core</u> <u>(n/cm² sec)</u>	
	<u>Maximum</u>	<u>Average</u>
Above 825 keV	1.5×10^{15}	0.97×10^{15}
Total	1.14×10^{16}	7.11×10^{15}
Peak Fuel Cladding nvt Above 825 keV	7.42×10^{22} n/cm ²	

residence time of two years the peak fuel cladding fluence (nvt) above 825 keV was determined to be 7.42×10^{22} neutrons/cm².

c. Nuclear Kinetics and Control

The major reactivity requirements for control rods in the LMFBR design are given in Table 2.15. These requirements satisfy the design safety criteria which were specified for the LMFBR Project Study⁽¹⁾.

The reactivity requirement for burnup is consistent with the changes in isotopic fuel composition during the 12-month refueling cycle. Tantalum rods are used for the burnup reactivity control and are positioned in the outer core region (Core Zone 2) to assist in controlling the radial power distribution (see Section 2.2.3.1-b). The choice of Tantalum shim rods meets the objectives of this study. The choice of B₄C may eventually prove to be more attractive. A discussion of B₄C rods is presented in the Task III report. Control for all other reactivity requirements is provided by boron carbide rods which are 25% enriched in B¹⁰.

Reactivity changes from cold shutdown to operation at full power (2400° F average fuel temperature and 950° F average coolant temperature) are based upon the design Doppler effect and the reactivity expansion coefficients. (The expansion coefficients are presented later in this section). The reactivity change from cold shutdown to the full power operating condition is made up of the following components:

	<u>Reactivity*(\$)</u>
Fuel Doppler Effect	-2.64
Radial Expansion	-1.57
Clad Axial Expansion	+0.08
Fuel Axial Expansion	-0.75
Sodium Thermal Expansion	+0.48
Total	-4.40

The axial expansion effects were based upon the expansion of fuel and tend to overpredict the negative reactivity contribution (i.e., clad restraints will reduce the axial expansion). In addition, a small positive reactivity effect due to sodium displacement by fuel radial expansion was neglected.

All primary and backup reactivity requirements were increased by 20% to provide a contingency allowance for design uncertainties.

The total number of control rods for the LMFBR design are given in Table 2.16. The average rod worths are consistent with the rod position and control rod bundle configuration shown in Figure 2.3.

The Doppler coefficient, Tdk/dT , for the two-zone, power-flattened core was calculated to be -0.0060 with sodium in and -0.0037 with sodium removed. Only the U-238 Doppler coefficient was assumed to give a significant contribution to the total Doppler effect. These Doppler coefficients are consistent with the safety criteria for the LMFBR Project study and reflect a conservative evaluation of new basic nuclear data which indicate larger capture-to-fission ratios for Pu-239. (The General Electric Standard data file will be presented and discussed in the final LMFBR Task III report.)

Power flattening reduces the Doppler coefficient at full power for a reactor of given design (including fuel rod dimensions), assuming that the peak fuel temperature is held just below the melting point at full power. This is because the average fuel temperature is higher in the power-flattened reactor than for the unflattened case with uniform fuel composition. Also, the average fissile composition is higher for the power-flattened core, yielding a slightly harder spectrum and hence, a lower Doppler coefficient.

$$* \beta_{\text{eff}} = 0.0038 \Delta k/k$$

TABLE 2.15

REACTIVITY CONTROL REQUIREMENTS

	Reactivity *	(\$)	
	Startup and Shim (Slow)	Scram (Fast)	Total
<u>Primary Control</u>			
Burnup (Equilibrium Refueling Cycle)	4.76	--	
Cold (208° F) to Hot Standby (810° F)	2.74	--	
Hot Standby to Full Power Operating	--	1.66	
Maximum Rod Worth (Stuck rod)	--	1.15	
Credible Sodium Void One Subassembly **	--	0.06	
Fuel Compaction One Subassembly ***	--	0.18	
Shutdown Margin	<u>--</u>	<u>1.50</u>	
Sub-Total	7.50	4.55	
Contingency (20%)	<u>1.50</u>	<u>0.91</u>	
Total Required	9.00	5.46	14.66
Total Available (Table 2.16)	10.85	5.95	16.80
<u>Backup Control</u>			
Cold (208° F) to Hot Standby (810° F)	--	2.74	
Hot Standby to Full Power Operating	--	1.66	
Credible Sodium Void One Subassembly **	--	0.06	
Fuel Compaction One Subassembly ***	--	0.18	
Shutdown Margin	<u>--</u>	<u>1.50</u>	
Sub-Total	--	6.14	
Contingency (20%)	<u>--</u>	<u>1.23</u>	
Total Required	--	7.37	7.37
Total Available (Table 2.16)	--	5.10****	5.10
Total Control Requirements	9.00	12.83	21.83
Total Control Available (Table 2.16)	10.85	11.05	21.90

* $\beta_{eff} = 0.0038 \Delta k/k$

** The worst voiding reactivity in one bundle

*** Compaction of the fuel from the upper and lower core into the central core region displacing all sodium, and the void regions in the upper and lower core replaced by axial blanket fuel.

****The apparent excess of requirements over the control available is acceptable since the part of the backup control for cold to hot standby (\$2.74) can be accommodated by slow control rod actuation.

TABLE 2.16

CONTROL ROD REQUIREMENTS AND AVERAGE ROD WORTHS

	Shim Rods (Tantalum)	Primary Scram Rods	Backup Scram Rods (25% enriched B ₄ C)	Startup Rods with Shim Drives**	Total
Number of Rods	12	7	6	6	31
Average Rod worth * (\$)	0.48	0.85	0.85	0.85	--
Total rod worth (\$)	5.75	5.95	5.10	5.10	21.90

* $\beta_{\text{eff}} = 0.0038 \Delta k/k$

** These Startup Rods, like the Scram Rods, are totally out of the core during power operation after achieving operating temperature.

The removal of sodium from the core and blankets of the LMFBR design results in a positive reactivity increment $\Delta k/k$ of 0.0085. Partial core voiding increases the positive reactivity effect to a maximum value of about 0.0198. The maximum reactivity occurs for a central void volume of approximately one-half of the core volume.

The safety criteria for the LMFBR design are based upon sodium voiding in one subassembly. The void reactivity increment for a center fuel bundle was determined to be 0.00023 or 6 μ .

Tables 2.17 and 2.18 list the temperature and power coefficients for the LMFBR design.

These coefficients were based upon the following material and geometric expansion effects:

Core Material Density Coefficients ($\rho dk/d\rho$)-
fixed core size

Fuel	0.3714
Sodium	-0.01397
Steel	-0.03357

Geometric Expansion Coefficient - fixed density

Hdk/dH	0.1953
Rdk/dR	0.1012

Doppler Coefficient (Tdk/dT)

Sodium In	-0.00600
Sodium Out	-0.0037

Net Effects

Axial Fuel Growth*	-0.1761
Radial Growth - Including Density**	-0.5816

* Hdk/dH - $\rho dk/d\rho$ (fuel)

** Rdk/dR - $2 \times \rho dk/d\rho$ (fuel, steel) + effects of added volume of sodium when core expands radially.

The prompt negative reactivity effect of the Doppler coefficient provides the main core safety factor. Unrestrained fuel axial expansion is also a prompt effect. However, it was conservatively assumed that the fuel growth is restrained by the cladding and for

TABLE 2.17

REACTIVITY TEMPERATURE COEFFICIENT

<u>Prompt</u>	$\Delta k / F \times 10^6^*$
Doppler	-2.40
<u>Delayed</u>	
Fuel Axial Expansion**	-1.98
Steel Axial Expansion	+0.38
Sodium Density Reduction	+0.90
Core Radial Expansion	-6.94
Total	-10.05
Reduction of Fuel and Steel Atom Density***	-7.61×10^{-6}
Increase of Sodium Volume Fraction***	-0.47×10^{-6}
Increase in Core Radius***	$+1.14 \times 10^{-6}$

* For a uniform 1° F temperature rise starting from full power, except for the Doppler effect which is for a 1° F rise of the average fuel temperature. The rise of the fuel temperature distribution followed the power distribution.

** Includes the geometric expansion of the core.

*** Components of the core radial expansion temperature coefficient.

the power coefficient the restrained fuel expansion effect is only ~25% of the unrestrained fuel expansion effect. The prompt negative power coefficient due to the Doppler effect was found to dominate the other delayed power coefficients.

The calculated neutron lifetime and effective delayed neutron fraction are as follows:

$$\ell = 0.34 \times 10^{-6} \text{ sec}$$

$$\beta_{\text{eff}} = 0.0038.$$

The principal delayed (long time constant) temperature coefficient of reactivity is that due to radial expansion of the core accompanying the heating of the support structure. Radial core expansion decreases reactivity primarily by decreasing the concentration of fuel in the core.

2.2.3.2 Reactor Thermal and Hydraulic Performance (F. W. Sciacca)

a. Thermal and Hydraulic Characteristics
Table 2.19 summarizes the reactor thermal and hydraulic data. The peak performance of

TABLE 2.18

REACTIVITY POWER COEFFICIENT

<u>Prompt</u>	<u>$(\Delta k/k)$ per $(\Delta P/P) \times 10^{3*}$</u>
Doppler Effect	-3.86
<u>Delayed</u>	
Fuel Axial Expansion **	-0.40
Steel Axial Expansion	+0.07
Sodium Density Reduction	+0.13
Core Radial Expansion	<u>-1.21</u>
Total	-5.27

* For a slow and small change of the power level near full power, a 1% increase of reactor power produces a 1.5° F increase in the average sodium temperature, a 1.5° F increase in the average structure temperature, a 2.0° F increase in the average cladding temperature and a 16.2° F increase in the average fuel temperature.

** Includes the geometric expansion of the core.

the reactor is limited by the two following criteria:

1. The fuel temperature must not exceed its melting point (5206° F) at any location in any fuel pin, and
2. The local fuel cladding strain must not exceed the local allowable strain at any point on any fuel pin.

The first criterion, fuel melting, limits the reactor power by determining the maximum local pin linear power, for a given fuel pin design and a given local cladding surface temperature. The peaking factors which determine the associated core average linear power are tabulated in Table 2.19. A discussion of the significance of each of these factors was presented in the Task I report. The resulting ratio of peak local power to core average power is 1.86.

The second criterion, local cladding strain, limits the reactor coolant outlet temperature by determining the maximum local coolant stream temperature for a given reactor coolant temperature rise. The peaking factors which determine the associated core average coolant enthalpy rise are tabulated in Table 2.19.

The mid-cycle radial and axial power profiles associated with the reference core design are shown in Figures 2.22 and 2.23 respectively. The core consists of two radial enrichment zones; hence the break in the radial power profile. The enrichment of the zone is planned such that the peak power in each zone is approximately the same.

The relative radial positions of the consecutive hexagonal rings of fuel assemblies are also indicated in Figure 2.22. There are

TABLE 2.19

SUMMARY OF REACTOR THERMAL
AND HYDRAULIC CHARACTERISTICS

Mid-Cycle Power Output, MWt			
Core			2118
Axial Blankets			183
Radial Blanket			<u>116</u>
	Total		2417
Core Power Peaking Factors			
Axial Power Distribution			1.20
Radial Power Distribution			1.22
Control Rod Flux Depression			1.05
Beginning of Life to Middle of Life Power Shift			1.05
Hot Spot (Uncertainties)			1.05
Overpower Allowance			<u>1.10</u>
	Cumulative Power Peaking Factor		1.86
Coolant Enthalpy Peaking Factors			
Radial Power Distribution			1.22
Control Rod Flux Depression			1.05
Beginning of Life to Middle of Life Power Shift			0.98
Subassembly Flow Distribution Effect			1.12
Effect of Orificing			0.90
Hot Stream (Uncertainties)			1.06
Coolant Dilution (Leakage and Radial Flow)			<u>1.05</u>
	Cumulative Coolant Enthalpy Peaking Factor		1.41
	<u>Core Average</u>	<u>Maximum* Fuel Temperature Location</u>	<u>Minimum Cladding Strain Margin Location</u>
Linear Power, kW/ft	9.63	17.9	13.5
Surface Heat Flux, MBtu/h-ft ²	0.502	0.93	0.70
Maximum Fuel Temperature, °F	--	5100	4400
Maximum Cladding Temperature, (mean radius) °F	--	1175	1225
Allowable Cladding Strain **	--	~3.3	~3.0
Cladding Strain Due to Fuel Swelling **	--	~3.2	~3.0

* At 10% overpower conditions

** At end of life conditions

FIGURE 2.22 RADIAL POWER PROFILE

FIGURE 2.23 AXIAL POWER PROFILE

seven such radial groupings in the inner core zone and three in the outer zone. The radial blanket region is composed of two radial groupings.

The overall core pressure drop is 64.3 psi. This is based on a total coolant flow of 80.6×10^6 lb/hr, with a net temperature rise of 340° F.

Flow through the radial blanket region is approximately one-tenth of the total. This amount is twice the power production fraction of the radial blanket. High flow is required in this region to compensate for the increase of power with lifetime which a given bundle experiences. Also the high radial power peaking in this blanket requires a large flow fraction relative to its power output. This is necessary in order to prevent overheating of the cladding. The bulk of the radial blanket is, in general, overcooled. The average coolant outlet temperature of this region is considerably below the mixed mean outlet for the core. This coolant temperature degradation is accounted for in arriving at the core heat balance.

Orifices are provided at either the inlet or outlet of each bundle to distribute the coolant flow according to the power profile, at least as far as this is possible. Orificing of a given bundle is such that the highest cladding temperature reached on any of the fuel pins within that subassembly is the same as the peak cladding temperature of the hot pin within the highest power subassembly. Thus the orificing is designed to limit cladding temperatures rather than to produce a uniform coolant outlet temperature from each subassembly.

This criterion for orificing makes it impossible to achieve a uniform coolant outlet temperature from all of the fuel subassemblies. As can be seen from Figure 2.22 the power falls off sharply with distance from the center of the core in the outer enrichment zone. In a single fuel assembly in this region

the pins nearer the core center produce considerably more power than do the fuel pins near the core edge. Thus the hottest pins in these assemblies dictate the minimal flow which the orificing can be designed for. Consequently, the bundle as a whole is somewhat overcooled. Its outlet temperature will be less than the average for the core. This also dictates that those fuel assemblies with a more uniform power distribution will have an outlet temperature higher than the core average.

With the above orificing criterion the fuel assemblies producing the highest core power, those in the core center, receive about 1.1 times the average bundle flow. This is also true of the inner row of fuel assemblies of the outer enrichment zone. Thus the peak channel coolant enthalpy rise is 1/1.1 (or 0.90) times what it would be if no orificing were used. This is the meaning of the effect of orificing peaking factor as shown in Table 2.19. Orificing produces the enthalpy peaking.

A histogram of temperature rise for each row of fuel assemblies is shown in Figure 2.24. Temperature rise is plotted against the square of radial distance from the core center. In this way the figure also gives an indication of the fraction of total core area which is occupied by a given row of fuel assemblies. Each row of fuel assemblies is composed of the fuel assemblies which form a given hexagonal ring about the core center. Because this is a hexagonal ring rather than a circular ring, certain fuel assemblies in each ring are closer to the core center than others in this same ring. Those assemblies nearer the core center will produce slightly higher power than those farther away. Thus the coolant outlet temperatures from the fuel assemblies in a given row will not be entirely uniform. The temperature rise shown in Figure 2.24 represents a mixed mean outlet condition for the respective rows of fuel assemblies.

The weighted average temperature rise of the core and axial blankets is 360° F. Figure 2.24 shows that the mean temperature rise of

FIGURE 2.24 CORE TEMPERATURE RISE - RADIAL PROFILE

most of the channel types exceeds this average. Only the two outermost rings of fuel assemblies have an average temperature rise which is lower than the core average. Note that 360°F is the temperature rise of the core and axial blanket region only. The mixed mean temperature rise of the core and radial blanket region together is 340°F. The degradation of core region outlet temperature is caused by the low temperature rise of the coolant flow through the radial blanket. These average temperatures are indicated in Figure 2.24.

Figure 2.25 is a temperature profile of the hot pin at overpower conditions. The highest fuel temperature occurs at the core mid-plane, approaching the melting point of 5206°F. The coolant temperature profile shown applies to the "hot stream." This combination of hot pin and hot stream is the most severe in terms of limiting the core thermal and hydraulic performance. The highest fuel temperature in the core is calculated based on the assumption that a central void is formed in the fuel. The void is formed due to the sintering and densification of fuel above a temperature of 1800°C. With these assumptions the initial size of the central void at core midplane in the hottest pin is calculated to be approximately 0.06 inches in diameter.

The reference fuel pin design is of the vented type. Fission gases are allowed to escape from the pin and there is no net internal pressure buildup. The choice of vented-to-coolant fuel meets the objectives of this study. Another means for handling the fission gases may eventually prove to be more attractive. The Task III report presents a discussion of sealed-pin fuel.

Cladding stresses due to gas pressure, therefore, do not enter into the cladding design criterion. The limiting design conditions of the cladding are those imposed by strains rather than by stresses. Cladding strain in

this vented pin design is caused by the action of fuel swelling.

The allowable cladding strain at a given point along a fuel pin is determined by the local mid-wall temperature and the local irradiation dose (fluence) of the cladding. Both the temperature and the fluence vary with axial position. Consequently, the allowable cladding strain varies continuously along the fuel pin. Similarly, the cladding strain caused by fuel swelling varies axially. It tends to follow the axial power shape of the pin. Thus the allowable strain and the expected strain change from point to point.

The design limiting location in the core is the position at which the margin between allowable cladding strain as established by local temperature and fluence levels and the cladding strain due to fuel swelling becomes a minimum. The condition occurs in the "hot pin" of the core. Figure 2.26 demonstrates how this margin varies axially along the hot pin. The most limiting point occurs at a location about 7/10 of the core length from the bottom of the core. It is here that the ratio of allowable strain to predicted fuel-swelling strain approaches unity. At all other locations this ratio is greater than one. The fuel pin approaches this condition only near the end-of-life when its burnup is at a maximum.

Included in Figure 2.26 is a plot of the mean (mid-wall) cladding temperature used in establishing the allowable cladding strain. This temperature is based on full-power operation at end-of-cycle, not overpower conditions. Because of the non-uniformity of the coolant flow area associated with a given fuel pin, circumferential temperature variations may occur around it. The presence of the wire spacer may increase this type of circumferential effect. Because of these possibilities an estimated additional 35°F of temperature peaking may occur. This peaking has been added to the cladding design temperature, and is included in the plot shown in Figure

FIGURE 2.25 AXIAL TEMPERATURE PROFILE

FIGURE 2.26 STRAIN MARGIN AS A FUNCTION OF AXIAL POSITION

2.26. The resultant design peak cladding temperature at the most limiting position is 1225°F.

The calculated corrosion allowance for the o.d. of the cladding at the most limiting position is less than one mil. Because the expected material removal from the cladding occurs over the lifetime of the fuel it is believed to have a very small influence on the predicted fuel swelling. Therefore, corrosion thinning of the cladding was not factored into the core design.

Other pertinent data applicable to the design limiting point are as follows:

Local Design Burnup, MWd/Te	138,000
Local Cladding Fluence (above 1 MeV)*	6.0×10^{22}
Design allowable cladding strain, %	3.0

* Note that the data on Tables 1.1 and 2.26 apply to the peak flux point and include the flux above 0.825 MeV.

These conditions occur in the hot pin at the end of its life in the reactor. A discussion of the peaking factors, assumptions, and models used in performing the thermal-hydraulic analysis are presented in Section III of the Task III report. The pressure drop correlation used throughout is as presented in Appendix B.2.2 of the Task I report. (3)

The present core design uses cladding temperatures based on the enthalpy peaking and power peaking factors in Table 2.9. An additional 35°F of cladding temperature peaking were allowed due to further uncertainties. However, the resultant cladding temperature estimates may not be conservative. If future tests show that the temperature peaking may be greater than that accounted for in the present design, several of the core design parameters may have to be changed.

b. Summary Fuel Fabrication Specification

The following specifications are typical of the requirements believed necessary in order to be reasonably assured of predictable fuel operation. The basis for the dimensional tolerances and the fuel homogeneity specification is uncertain. The values shown are approximate.

Cladding

1. All tubing shall be manufactured and supplied in conformance with ASTM designation A213-65, "Seamless Ferritic and Austenitic Alloy Steel Boiler, Superheater, and Heat Exchanger Tubes" and with the supplemental requirements set forth in this specification.
2. The internal and external surfaces shall be clean and bright, free from discoloration, grease, dirt, metal particles and other foreign matter. Maximum surface roughness shall be equivalent to ASA designation 32 on either i.d. or o.d.
3. ASTM grain size 5-8 as determined by the current edition of ASTM-E112.
4. All finished tubing procured to this specification shall be ultrasonically tested. Discontinuities, in depth within the tubing up to and including 10% of wall thickness, shall be deemed acceptable.
5. In order to meet the most severe reactor conditions of 3.0% strain in cladding with an average temperature of 1225°F and a burnup of 138,000 MWd/Te, the cladding shall be capable of a minimum uniform elongation of 3.0% at strain rates between 10^{-2} and 10^{-8} (1/min) while being irradiated to a total fluence (>1 MeV) of 6.0×10^{22} nvt.
6. The tubing might be a microstructurally modified form of 316 or 304 stainless steel.

7. Tubing dimensions shall be the following:

O. D. = 0.248 in. $\pm$ 0.001 in.
 I. D. = 0.228 in. $\pm$ 0.001 in.
 Variations in tube wall thickness
 must not exceed 0.001 in. at any
 cross section.

Fuel

1. The fuel material shall be mixed oxide pellets of plutonium and uranium.
2. The pellet o.d. shall be 0.222 in. $\pm$ 0.002 in.
3. The nominal density of the pellet shall be 89.6% $\pm$ 1.0% of the theoretical maximum. The nominal pellet to cladding radial gap is 0.003 in. The resulting smeared density at power is 85%.
4. The atomic ratio of oxygen to metal in the fuel pellets shall be 1.98 $\pm$ 0.01.
5. The homogeneity of the fuel pellets shall be such that no plutonium-rich particle (Pu concentration greater than 1.5 times the average) exists which has a size greater than 60 microns.
6. After fabrication the fuel pin shall be evacuated and back filled with helium to an absolute pressure of 1 atm.

Axial Blanket

1. The blanket fuel is depleted uranium dioxide.
2. The pellet density shall be 95% $\pm$ 2% of the theoretical maximum.
3. The pellet o.d. shall be 0.224 in. $\pm$ 0.002 in.
4. The upper and lower blanket height shall be 15 in. $\pm$ 0.15 in.

Radial Blanket

1. The blanket fuel is depleted uranium dioxide.
2. The fuel pellet density = 95% $\pm$ 2%.
3. The cladding material is to be the same as that used in the core region.
4. The dimensions of the fuel cladding shall be the following:

Tube o.d. = 0.418 in. $\pm$ 0.001 in.
 Tube i.d. = 0.388 in. $\pm$ 0.001 in.
 Variations in tube wall thickness
 must not exceed 0.001 in. at any
 cross section

5. The blanket height shall be 60 in. $\pm$ 0.002 in.
6. The pellet o.d. shall be 0.382 in. $\pm$ 0.002 in.

Welding

1. Welds shall be fully inspected (both x-ray and helium leak check).
2. Weld penetration shall not be less than 90%.
3. Cladding shall be free of welds at any point in the fuel region, such as welding a spacer to the cladding, etc.

Wire Wrap Spacer

The wire shall be of the same material as the cladding.

The tolerances specified above for the core and blankets are consistent with present technology and capabilities. There are incentives to reduce the tolerances if this can be done economically. At present, closer tolerances can be held on fuel pellet diameter, but only by resorting to an expensive grinding procedure.

c. Fuel Pin Vent Operation (F. W. Sciacca)

The fuel pin venting device is designed to allow the escape of fission gases as they build up within the pin, while preventing the entrance of sodium into the fuel region of the pin. The latter condition must be satisfied under any situation that the pin may be subjected to during its life. The gas space in the pin is filled with helium during fabrication and the pin is then sealed by means of soft solder. The seal prevents air from entering the pin until the solder is melted. This occurs only after the fuel assembly is placed in the reactor core. Once the vent tube is opened, gases can escape from the fuel pin and sodium can enter into the vent cup of the venting device. The sodium level in the vent cup is affected by the operating temperature within the pin and by the pressure at the vent hole. During normal operation the pressure at the vent holes is on the order of 28 psia. Under zero flow conditions with the bundle in the core the value is about 19 psia. Out of core, of course, the pressure is assumed to be ambient, here taken to be about 14.7 psia.

During normal operation fission gases are continually generated within the fuel pin. This prevents the rise of the sodium level in the vent cup, and it remains only as high as the bottom of the vent tube. Table 2.20 describes the various conditions to which the fuel pins will normally be exposed during their lifetime. The sodium level in the vent cup remains at acceptable levels under all of these conditions. This is true of both normal pins and possible "hot" pins. The sodium level in the vent device is described in Table 2.20. The hot pin is the most critical from the standpoint of pulling in sodium when the gas cools and contracts. The highest sodium level in the vent cup has been calculated to be about 9 inches above the bottom of the vent tube. Since the total active cup length is about 17.5 inches, there is considerable margin available before sodium

could possibly get into the fuel region of the fuel pin.

Table 2.20 also indicates the condition at which gas will escape from the fuel pins during refueling operations. This condition applies to fuel assemblies which are being removed from the core at end of life. During this operation it is desirable to prevent fission gases within the pins from getting into the hot cell. Whether or not these gases escape from the pins depends on the gas temperature. For a pin classified as hot, gases will escape from the vent when the temperature exceeds about 600° F. For an average pin this escape temperature is only about 500° F. The fuel transfer cooling apparatus has the capability of keeping the temperature below the indicated levels.

Details of the vent device are shown in Figure 2.1.

2.2.4 Reactor Instrumentation (N.W. Brown, D. M. Switick)

A large economic incentive exists for an LMFBR system which employs a core with optimum fuel geometry. To accomplish this, plant safety relies on engineered safeguards rather than inherent physical properties of the core design to maintain the public and plant safety. This is necessary because of the large energy release estimated for disassembly accidents associated with the advanced core designs.

It is proposed that a system of core instrumentation can reduce the probability of malfunctions propagating into a DBA to such an extent that the public and plant safety is assured. (This probably would be on the same order of probability for a pressure vessel failure in a water reactor).

The extent of instrumentation presently believed to be required is described in this section. The Task III and IV report will dis-

TABLE 2.20

FUEL PIN VENT OPERATION

Gas Temperatures, ° F

	Core Region	Axial Blkt. Region	Vent Region	Pressure at Vent Hole, psia	Sodium Level* in Vent Cup, in.
Initially placed in core before startup-time a)** of vent seal melting b)**	400 (400) 400 (400)	400 (400) 400 (400)	400 (400) 400 (400)	19 (19) 25 (25)	0.0 (0.0) 0.9 (0.9)
Normal operation	3200 (2400)	1310 (1200)	1300 (1175)	28	0.0 (0.0)
10% Overpower Operation	3450	1330	1300	28.0	0.0
Cool down to refueling temperature—in core	400 (400)	400 (400)	400 (400)	19	9.15 (7.7)
Cool down to refueling temperature —removed from core	400 (400)	400 (400)	400 (400)	14.7	4.6 1.4
Zero power, out of core, conditions at which gases will escape from pin	590 (490)	590 (490)	590 (490)	14.7	0.0 (0.0)

Values apply to the hot pin. Values in parentheses apply to the average pin.

** 0.0 level is taken at bottom of vent tube

** a) No Na flow

** b) Full Na flow

cuss the rationale for this choice of safety instrumentation.

Major objectives for the core protection system are as follows:

1. Automatically initiate and complete appropriate protective action whenever the plant approaches an unsafe condition,
2. The probability of an unsafe failure in the protection system must be less than 10^{-9} , and
3. Plant unavailability due to instrument failure must be less than 5%.

The latter two objectives are the result of requiring the core instrumentation system to provide the same level of safety assurance that is available on today's reactor plants.

The protection system is intended to be designed in accordance with the IEEE Standard for the Minimum Functional Performance and Reliability of Nuclear Power Plant Protection Systems. The more important standards are:

1. No single failure in the system shall prevent proper protective action,
2. There shall be a high degree of separation between the plant safety and normal control functions, especially where a malfunction in one system would affect the other,
3. The protective function channels shall be completely independent and physical isolation shall be provided between channels which provide the same protective function,
4. An individual channel may be made inoperative for maintenance, calibration or test during power operation,

5. The system shall provide the operator with pertinent and timely information of its own status and plant safety, and
6. The design will facilitate the recognition, location, replacement, repair or adjustment of malfunctioning components.

Where exceptions to these standards may be contemplated, it is recognized that such exceptions must eventually be justified by detailed reliability analyses.

2.2.4.1 Core Instrumentation Requirements

A schematic presentation of the extent of instrumentation believed to be required is given in Figure 2.27. The present requirements are for 1623 independent protective function channels.

Safety system functions are provided wherever possible with multiple redundancy in all components and operate on a two out of three logic scheme to trip the system. Except as noted, an operational spare channel is incorporated to enhance reactor plant availability. The notable exception to this is the subassembly flow measurement, which, as shown in Figure 2.28, depends on a single sensor. One could add an additional flow sensor, or sensors, at the outlet of each assembly. The actual requirement, as well as the final choice of the type of flowmeter to be used, will require a comprehensive study of the desired characteristics as compared with the current technology for the various alternatives, followed by development, and then by testing to obtain reliability data.

The system is accompanied by an automatic testing sequence to test each protective function channel. Besides checking the response of the circuitry, a means is also provided to test the sensor elements.

FIGURE 2.27 SCHEMATIC OF PROTECTIVE FUNCTION CHANNELS PER BUS
(THREE REDUNDANT BUSES IN SAFETY SYSTEM)

Core protection is based on having adequate analysis and prior testing to diagnose the total core and subassembly conditions (including knowledge of the damage propagation mechanisms and related time scales) as measured by the following detectors and instrumentation:

1. For each fuel subassembly, the flow and outlet temperature are measured. Figure 2.28 depicts the functional arrangement of this instrumentation. At some signal level, either flow or temperature alone will result in a scram. For indications of less severe deviations, flow and temperature coincidence is required.
2. Sodium temperature is measured at the outlet of each primary pump and at the inlet of each intermediate heat exchanger (IHx). The functional integration of those sensors into the safety system is similar to the thermocouple integration in Figure 2.28.
3. Three seismic instruments monitor earth crustal movements. (An operational spare is not required.)

4. Sodium level monitors are located in the main tank. The safety system is tripped on either low or high sodium levels.

5. Neutron flux is measured as leakage flux just exterior to the reactor vessel. Several ranges of monitoring are available:

- a. Source (start-up)
- b. Intermediate
- c. Wide
- d. Power level

The primary neutron detector in each range is an ion chamber which is gamma compensated for 5 (b) and 5 (c). The integration of the detectors is shown in Figure 2.29. Although shown in the figure, the power range monitors are not tied into the safety system but are used as part of the normal plant control system. A backup d.c. power supply is provided for each of the four flux monitor assemblies.

FIGURE 2.28 SODIUM COOLED FUEL BUNDLE SAFETY INSTRUMENTATION (SIMILAR TO GE DRAWING 277E204)

TABLE NO. 1 NEUTRON MONITORING SYSTEM OPERATIONAL RANGES

NEUTRON FLUX AT DETECTOR n/cm ² /sec. ±	10 ⁻²	10 ⁻¹	10 ⁰	10 ¹	10 ²	10 ³	10 ⁴	10 ⁵	10 ⁶	10 ⁷	10 ⁸	10 ⁹	10 ¹⁰
COUNTS PER SECOND (SRM)	.1	1	10	10 ²	10 ³	10 ⁴	10 ⁵	10 ⁶					
AMPERES (IRM, WRM, PRM)				3.16 × 10 ⁻¹²	3.16 × 10 ⁻¹¹	3.16 × 10 ⁻¹⁰	3.16 × 10 ⁻⁹	3.16 × 10 ⁻⁸	3.16 × 10 ⁻⁷	3.16 × 10 ⁻⁶	3.16 × 10 ⁻⁵	3.16 × 10 ⁻⁴	
FOR INFORMATION ONLY	ESTIMATED FULL POWER												
SOURCE RANGE MON.	[Bar chart showing range from 10 ⁻² to 10 ¹]												
INTERMEDIATE RANGE MON.	[Bar chart showing range from 10 ⁰ to 10 ⁴]												
WIDE RANGE MON.	[Bar chart showing range from 10 ² to 10 ⁷]												
POWER RANGE MON.	[Bar chart showing range from 10 ⁴ to 10 ¹⁰]												

* MIXED SPECTRUM EQUIVALENT TO BORON-10 THERMAL CAPTURE CROSS SECTION

FIGURE 2.29 NEUTRON MONITORING SYSTEM (SIMILAR TO GF DRAWING 277E254)

2.2.4.2 Safety System Actions

The reaction of the safety system to trips can be any one of the following:

1. Alarm,
2. Power limitation,
3. Run-in, or
4. Scram.

An alarm notifies the operator of a low priority trip and indicates the trip origin. The operator is solely responsible for any further action based on an alarm.

Power limitation is used to prevent an increase of the core power level. It is provided to prevent a marginal situation from degenerating into one requiring more drastic protective system action. The operator must manually clear a power limitation trip to increase power in the core.

Run-in is a higher level priority trip which provides for the insertion of all control rods at their normal speed until the operator terminates the action. A deliberate clearance must be made by the operator to again raise power and the initial causation for the trip must be remedied. This protective system reaction is employed to ease the thermal shock on the reactor system and is provided for those protective functions which do not require immediate action to protect the reactor.

A scram provides a rapid automatic insertion of all safety rods which drives the core subcritical. The safety rods cannot be removed until the cause for the trip can be positively identified and remedied.

2.2.4.3 Fuel Bundle Inlet Flow Measurement Device (J. H. Germer)

Figure 2.30 shows a device which measures variations in individual fuel channel flow rate. It incorporates six level sensing tanks located within the outlet plenum of the reactor (shown in Figure 2.6 and Figure 2.7). Small tubes connect to the venturis below the inlet to each fuel bundle (Figure 2.2). The decision to measure flow at the inlet of each fuel bundle meets the objectives of this study. Outlet flow measurements may eventually prove to be more attractive. A discussion of outlet flow measurement is presented in the Task IV report.

Each level sensing tank handles 44 or 45 individual level sensors. Each of the tubes which connect to the venturis terminates within a level sensing tank in a manner which permits the insertion of an electrical sodium level sensor. An argon pressure controller regulates the pressure of the argon cover gas in the level sensing tank to maintain liquid levels within the range of the electrical level sensors. A drain tube extends to near the bottom of the level sensing tank to remove any sodium which might accumulate in the bottom of the tank. The drain tube normally has a small outflow of argon gas. This continuous outflow of gas makes it possible for the argon pressure regulator to operate with a continuous flow in one direction only.

Fuel bundle orifices are located downstream of the flow venturis. The size of the venturis is varied so that they all provide approximately the same pressure indication. Any indicated difference in level by the sodium level sensors thus indicates a deviation from normal flow in a fuel bundle.

In order to assure that all of the connecting tubes are free from constriction, the argon pressure can be cycled continuously over a small range. This cycling causes

FIGURE 2.30 FLOW MEASURING DEVICE FOR FUEL BUNDLE

a periodic variation in the indications from all of the sensors. A decrease in amplitude of this variation on any one sensor is a positive indication that there is partial plugging of that tube. In that event, the system can be purged with sodium by temporarily inactivating that particular group of sensors and decreasing the argon pressure sufficiently to cause sodium to overflow the tops of the sensor tubes. The arrangement of the piping to the sensors is such that each sensing tank is connected with fuel bundles scattered over the entire core. If one level sensing tank is being purged at the time when a flow stoppage occurs, it is still possible to detect the accident after it propagates past a single fuel bundle.

The flow sensor detects the deviation of flow in one fuel bundle from the average flow in the core. An absolute measurement of flow is obtained by also measuring the argon pressure in the sensing tank.

It will be noted from Figure 2.30 that the sodium level sensors are built as a part of a cover that can be removed as a unit without severing any electrical leads.

2.2.4.4 Outlet Thermocouples (J. H. Germer)

Thermocouples are located over all of the fuel bundles except those 31 which contain control rods. This does not completely comply with the objective given in Section 2.2.4.1 which calls for thermocouples over all fuel bundles. As details of protective instrumentation become available from R&D efforts; it is anticipated that this inconsistency can be resolved.

The method of measurement consists of supplying probes extending below the shield plug and in line with each fuel bundle. Each probe is a single unit as shown schematically in Figure 2.31, and can

be separately disconnected and replaced from above the shield plug. Its upper end contains electrical leads which are plugged into a master connector at one side of the shield plug. This master connector permits removal of all connections in one operation before raising the shield plug for refueling. Also at the upper end of the probe are a static seal and a means of hold-down.

The probe fits over most of its length within a guide tube which permits installation and removal of the probe. This guide tube is a permanent part of the shield plug, and is guided at one or more levels against radial motion. By having the guide tube open at the lower end only, sodium may rise in the annulus around the tube and form a frozen seal in the shield plug region. A slight taper in the sides of the probe permits easy removal.

Each probe has four thermocouples. Three thermocouples are in use. One would consider a reading to be valid only if at least two thermocouples have similar indications. In the case of a malfunction of one of a group of thermocouples, the operator is alerted to take remedial measures. With the fourth thermocouple provided in each group as a spare, the operator can manually change the connection in the control room.

2.2.5 Nuclear Radiation Shielding (M. L. Weiss)

Nuclear radiation shielding is required to perform the following functions:

1. Limit the exposure of personnel which results from direct radiation from the reactor core and the primary sodium.

FIGURE 2.31 CORE OUTLET THERMOCOUPLES

2. Limit the exposure of personnel which results from direct radiation from a bare fuel assembly which is suspended in the refueling cell.
3. Limit the exposure to fast neutrons of critical structural parts of the core and the primary heat transfer system.
4. Limit the activation of the sodium in the secondary heat transfer system.
5. Limit the activation of the primary sodium pump and of the intermediate heat exchanger.

Features are incorporated into the designs of the reactor vessel, of the primary heat transfer system, and of the refueling cell to meet these requirements. The following text presents detailed criteria for the shielding designs and the results of a bulk shielding analysis of the effectiveness of the shielding.

2.2.5.1 Biological Shielding

a. Criteria:

The first concern in a nuclear radiation shield design is the protection of personnel. Maximum permissible radiation dosages to personnel in both restricted (controlled radiation) areas and in unrestricted areas are specified in 10 CFR 20. To provide a principal design basis for the dose rate criteria assumed for this shielding evaluation, the following principal limits are being observed:

1. Personnel exposure limits in restricted areas
Whole body—5 Rem/year
(The average dose rate assuming a 40-hour work week and a 50-week work year is 2.5 mRem/h.)

2. Personnel exposure limits in unrestricted areas
Whole body—0.5 Rem/year
(The average dose rate assuming a 40-hour work week and a 50-week work year is 0.25 mRem/h.)

b. Core Radial and Upper Axial Shielding

Figures 3.1 and 3.2 illustrate the core, tank and adjacent biological shielding. The principal sources of radiation which must be shielded include:

- Fission and neutron capture gammas,
- Fission neutrons,
- Fission product decay gammas, and
- Neutron activation gammas

Tables 2.21, 2.22, and 2.23 illustrate the radial and upper axial shielding compositions and thicknesses assumed for this conceptual shielding design analysis.

Results of bulk shielding type calculations, determined by using the removal-diffusion shielding program, MAC, are illustrated by Figures 2.32, 2.33, 2.34, and 2.35. The first two figures illustrate the total neutron flux and gamma dose rate respectively, as a function of radial distance from the center of the core. By extrapolating the curves of Figure 2.32 and applying a $1/r$ geometric attenuation correction, the approximate regular concrete wall thickness required to reduce the radial neutron dose rate to 1 mRem/h was determined to be 11 feet. This thickness is applicable where the core is closest to the tank wall. At the other extreme, only 4.5 feet of regular concrete is required for neutron shielding.

However, the gamma shielding requirement must also be considered. Figure 2.33 indicates, by extrapolation and $1/r$ geometric attenuation, that a regular concrete wall thickness of 10 feet is

TABLE 2.21

MATERIAL COMPOSITIONS FOR THE RADIAL SHIELD (FOR IHX)

(Measured radially outward from the center of the core through the center of the nearest intermediate heat exchanger.)

Region	Region Width (cm)	Outer Radius (cm)	Material	Volume Fraction	Density ₃ (gm/cm ³)
Core	107.12	107.12	Fuel	0.467	9.35
			Sodium	0.375	0.8512
			Stainless Steel	0.149	7.93
			Tantalum	0.004	15.80
Blanket	47.06	154.18	Fuel	0.518	9.35
			Sodium	0.277	0.8512
			Stainless Steel	0.205	7.93
Reflector	13.25	167.43	Stainless Steel	0.927	7.93
			Sodium	0.073	0.8512
Interim Fuel Storage	29.00	196.43	Stainless Steel	0.200	7.93
			Sodium	0.80	0.8512
Neutron Shield	44.20	240.63	Stainless Steel	0.800	7.93
			Sodium	0.200	0.8512
Sodium Pool	374.90	615.53	Sodium	1.00	0.8512
Boron Carbide*	4.20	619.73	Boron Carbide	0.606	1.85
			Stainless Steel	0.175	7.93
			Sodium	0.219	0.8512
Steel Plates	7.62	627.35	Stainless Steel	1.00	7.93
Inner IHX	200.00	827.35	Stainless Steel	0.10	7.93
			Sodium	0.90	0.8512

* An alternate with the boron carbide region replaced by a sodium region was also evaluated.

TABLE 2.22

MATERIAL COMPOSITIONS FOR THE RADIAL SHIELD (FOR TANK WALL)

(Measured radially outward from the center of the core through the point at which the core is nearest to the tank wall.)

Region	Region Width (cm)	Outer Radius (cm)	Material	Volume Fraction	Density ₃ (gm/cm ³)
Core	107.12	107.12	Fuel	0.467	9.35
			Sodium	0.375	0.8512
			Stainless Steel	0.149	7.93
			Tantalum	0.004	15.80
Blanket	47.06	154.18	Fuel	0.518	9.35
			Sodium	0.277	0.8512
			Stainless Steel	0.205	7.93
Reflector	13.25	167.43	Stainless Steel	0.927	9.35
			Sodium	0.073	0.8512

TABLE 2-22 (Con't)

Interim Fuel Storage	29.00	196.43	Sodium	1.00	0.8512
Neutron Shield	53.34	249.77	Stainless Steel	0.800	7.93
			Sodium	0.200	0.8512
Sodium Pool	85.51	335.28	Sodium	1.00	0.8512
Primary Tank	5.08	340.36	Stainless Steel	1.00	7.93
Thermal Insulation	15.25	355.61	Stainless Steel	1.00	0.80
Outer Tank Plus Liner	3.81	359.42	Stainless Steel	1.00	7.93
Biological Shield	365.76	725.18	Regular Concrete	1.00	2.20

TABLE 2.23

MATERIAL COMPOSITIONS ABOVE CORE MIDPLANE

Region	Region Thickness (cm)	Accumulated Thickness (cm)	Material	Volume Fraction	Density ³ (gm/cm ³)
Core	38.10	38.10	Fuel	0.467	9.35
			Sodium	0.375	0.8512
			Stainless Steel	0.149	7.93
			Tantalum	0.004	15.80
Axial Blanket	38.10	76.20	Fuel	0.467	9.35
			Sodium	0.374	0.8512
			Stainless Steel	0.149	7.93
			Tantalum	0.005	15.80
Gas Plenum	53.34	129.54	Sodium	0.375	0.8512
			Stainless Steel	0.149	7.93
			Void	0.476	0.00
Handles	17.78	147.32	Stainless Steel	0.10	7.93
			Sodium	0.90	0.8512
Sodium Pool	300.00	447.32	Sodium	1.00	0.8512
Steel Plates	30.48	477.80	Stainless Steel	1.00	7.93
Top Shield Plug	121.92	599.72	Boron Carbide	0.02	1.85
			Carbon	0.70	1.60
			Iron	0.28	7.7

FIGURE 2.32 TOTAL NEUTRON FLUX VERSUS RADIAL DISTANCE FROM CENTER OF CORE

FIGURE 2.33 GAMMA DOSE RATE* VERSUS RADIAL DISTANCE FROM CENTER OF CORE

FIGURE 2.34 TOTAL AND THERMAL NEUTRON FLUX VERSUS HEIGHT ABOVE CORE MIDPLANE

FIGURE 2.35 GAMMA DOSE RATE* VERSUS HEIGHT ABOVE CORE MIDPLANE

required to reduce the gamma dose rate to 1 mR/h where the core is closest to the tank wall. At the other extreme, where the core is farthest from the tank wall, fission and neutron capture gammas require 5 feet of regular concrete, whereas Na-24 decay gammas require 6 feet of regular concrete shielding. This preliminary analysis indicates 11 feet of regular concrete biological shielding is required where the core is located closest to the tank, whereas, 180 degrees away, this thickness may be reduced to 6 feet.

In the upper axial direction, Figures 2.34 and 2.35 indicate that the 1-foot-thick steel plates (thermal shield) plus the 4-foot-thick top shield plug will reduce the dose rate to about 2 mRem/h (combined neutron and gamma radiation). Actually, with a geometric attenuation factor applied, the dose rate would be only a fraction of a mRem per hour. Such a radiation level would permit 40-hour per week access into the refueling cell during full power operation. However, the design objective is to permit refueling cell access 1 day after reactor shutdown. In comparing the calculated radiation level, then, to the design objective, it would appear that the top head shielding is in excess of needs. However, this calculation does not include the effects of penetrations nor cooling passages, etc. Hence, the allowance of a 4-foot thickness for the top head shield probably should be retained until details of penetrations are evaluated.

c. Refueling Cell Wall Thickness

The estimated shielding mass requirement for the refueling cell walls is 425 gm/cm^2 . The shielding is expected to result in a peak dose rate in the adjacent space of about 5 mR/h from an unshielded fuel assembly. A shutdown time of 1 day is assumed. This dose rate is twice the average level which personnel

are permitted to receive during a 40-hour work week. However, refueling operations are expected to occur no more than twice annually (preferably only once); further, the duration of actual fuel transfer operations is estimated to require considerably less than 5 days; and, finally, the actual time a fuel bundle is in a location that it could cause a dose rate of 5 mR/h in an adjacent space is estimated to be well under 50% of its transfer time.

The 425 gm/cm^2 shielding requirement is equivalent to the following thicknesses of possible gamma shielding materials:

<u>Shielding Material</u>	<u>Density (gm/cm³)</u>	<u>Thickness (feet)</u>
Regular Concrete	2.35	6.0
Magnetite Concrete	3.5	4.0
Iron	7.7	1.8

Four feet of magnetite concrete was selected for the refueling cell construction.

2.2.5.2 Fast Neutron Fluence

a. Criteria

The design limit for fast neutron fluence ($E_n > 1 \text{ MeV}$) at permanent austenitic stainless steel components was taken to be $1 \times 10^{21} \text{ neutrons/cm}^2$.

Permanent components are expected to fulfill functional requirements during the lifetime of this plant which is estimated to be 35 years. A load factor of 0.80 is assumed.

b. Primary Sodium Tank

Figure 2.36 illustrates the calculated fast neutron fluence radially outward from the core. The primary sodium tank is located

FIGURE 2.36 FLUENCE, ($E_n > 1.0$ MeV), VERSUS RADIAL DISTANCE FROM CENTER OF CORE

11 feet from the center of the core at the closest point. Figure 2.36 indicates that the fast neutron fluence at this point is 2×10^{17} neutrons/cm² ($E_n > 1$ MeV), which is more than a factor of 1,000 less than the assumed criterion. With this margin, fast neutron irradiation damage to the tank wall is not expected to be a problem.

The total neutron flux at this location was calculated to be 2×10^{12} neutrons/cm²-sec (Figure 2.32), resulting in a lifetime total neutron fluence of 1.8×10^{21} neutrons/cm². The latter number is provided, in addition to the greater than 1 MeV neutron fluence, as a reference for comparison to new or alternate neutron fluence limits or standards which may become important in the future.

c. Reactor Vessel

Indicated as a part of the neutron shield on Figure 2.36 is a cylindrical vessel located about 4 inches in from the periphery of the neutron shield. This component is the reactor vessel. It is assumed to be a nonreplaceable component. Being located so near the core, neutron irradiation damage to the reactor vessel is a concern. Figure 2.36 indicates that the fast neutron ($E_n > 1$ MeV) fluence at this location is expected to reach about 2×10^{19} neutrons/cm² over the plant lifetime. This fluence is a factor of 1/50 times the assumed design limit. The total neutron fluence at the reactor vessel is estimated from Figure 2.32 to reach 1.7×10^{23} neutrons/cm² over the life of the plant.

d. Grid Plate

In the lower axial direction, the grid plate is located only about 2.5 feet below the blanket. The material composition of the shielding and structure separating the blanket and grid plate will be, approx-

imately, 25 volume percent stainless steel and 75 volume percent sodium. At full power, the fast neutron ($E_n > 1$ MeV) flux at the lower edge of the blanket will be about 3.4×10^{13} neutrons/cm²-sec. Assuming a 35 year plant lifetime, 0.80 load factor, and fast neutron relaxation lengths of 6.1 and 33 centimeters, respectively, for stainless steel and sodium, the calculated fluence ($E_n > 1$ MeV) at the grid plate will be about 7×10^{20} neutrons/cm², which is just under the assumed design limit.

2.2.5.3 Activation

a. Criteria

The activity of the secondary sodium coolant should be limited in order that shielding of the secondary system is not required. An approximate nuclear radiation dose rate of 7.5 mR/h three feet away from a large sodium pipe is considered to be a reasonable design limit.

It is assumed that some repair or maintenance work on the primary sodium pumps and intermediate heat exchangers will be necessary at one or more times during the lifetime of this plant. Accordingly, it is necessary to establish a design limit for the radiation level, due to neutron activation, near the surface of the pumps and heat exchangers. A limit of 100 mR/h is considered to be reasonable for this radiation source.

b. Primary Sodium Activity

Activation of the primary coolant is of interest for the following reasons:

1. After reactor shutdown, one of the principal radioactive sources which must be shielded is the Na-24 decay activity in the large tank.

2. In addition to Na-24 with its characteristic 15-hour half-life, there is a very significant amount of radioactivity in the primary coolant due to the (n, 2n) reaction in and near the core which produces Na-22. The latter radioisotope has a very long half-life (2.6 years), and becomes the principal contributor of coolant radioactivity about 10 days after reactor shutdown. It is the Na-22 activity which will create the principal radiation hazard from sodium surface films on components which must undergo maintenance or service operations.
3. The Na-24 gamma energy emission may be sufficient to create excessive gamma ray heat generation in the concrete biological shield which surrounds the primary sodium tank. At any rate, the magnitude of such heating must be evaluated.

Calculations of Na-24 activity indicate a full power equilibrium decay rate of 5.410×10^8 disintegrations/cm³-sec, or 0.0146 Ci/cm³ (assuming a primary coolant volume of 55,000 ft³, or 1.557×10^9 cm³). This activity results in a dose rate of 49,300 or approximately 50,000 R/h at the surface of a large container of this coolant. As shown by Figure 2.35 this dose rate is not significant, at the surface of the pool during full power operation, when compared to the gamma dose rate from core fission gamma and neutron capture gammas. The latter sources give a dose rate of nearly 10 million R/h at the surface of the pool.

An approximate calculation of the Na-22 activity indicates a full power equilibrium decay rate of 2.290×10^5 disintegrations/cm³-sec, or 6.19 μ Ci/cm³, again assuming a primary coolant volume of 55,000 ft³, or 1.557×10^9 cm³. This

activity results in a dose rate of 9.0 R/h at the surface of a large container of this coolant. Of course, during operation and for the first few days after reactor shutdown, this dose rate due to Na-22 decay is not significant when compared to the Na-24 decay. However, the Na-22 activity becomes predominant 10 days after reactor shutdown.

Figures 2.33 and 2.35 illustrate the attenuation of Na-24 decay gammas by concrete and steel, respectively. These figures and other calculations indicate the following thicknesses of gamma shielding materials will reduce the 50,000 R/h dose rate at the pool surface to 1 mR/h:

Shielding Material	Density (gm/cm ³)	Thickness (feet)
Regular Concrete	2.35	6.5
Magnetite Concrete	3.5	4.4
Iron	7.8	2.0

One foot of iron and 2.2 feet of magnetite concrete are provided in the reactor shield plug.

c. Secondary Sodium Activity

Activation of the secondary sodium coolant is of major interest because it would be difficult to shield the secondary system. Further, it is the secondary sodium which transfers its heat to generate steam in the plant boilers. A water-sodium reaction involving radioactive sodium presents a greater hazard than such a reaction involving nonradioactive sodium.

A permanent thermal neutron shield (not shown on the figures) is provided on the side facing the reactor of the intermediate heat exchangers and at the primary sodium pumps. This shield consists

of two closely packed rows of 1-inch vertical rods. The rods contain natural boron carbide (B_4C) at 75% of its theoretical maximum density. The rods are clad with stainless steel tubes which are hermetically sealed. The helium which is generated is collected in a plenum at the bottom of the rod. The expected burnup of the rods is less than 3.5 atom percent of the B^{10} over the 35-year design life.

For an unshielded secondary coolant system, the design criteria indicates a dose rate limit of about 7.5 mR/h three feet away from a large sodium pipe. An approximate calculation, assuming the geometry of Figures 3.1 and 3.2, and a secondary coolant loop volume of 3,865 cubic feet, indicates a dose rate three feet away from the surface of a 30-inch diameter pipe to be 20 mR/h without the addition of neutron shielding near the heat exchanger. The activity in the sodium is approximately 0.04 $\mu\text{Ci/cc}$.

The effect of the boron carbide shield (Table 2.21) was evaluated using the same approximate calculational method. This shielding reduces the dose rate to 0.03 mR/h and reduces the sodium activity to 6×10^{-5} $\mu\text{Ci/cc}$. This is a factor of about 700. Figure 2.37 indicates the effect of the boron carbide shielding on the thermal neutron flux in the location of the intermediate heat exchanger. The thermal neutron flux is the principal contributor to secondary sodium activation inside the heat exchanger.

d. Pump and Heat Exchanger Activation

Preliminary estimates, based on the neutron flux distributions illustrated by Figures 2.32 and 2.37 indicate the following dose rates at the surface of the pumps and the heat exchangers which are located

nearest to the core (Figure 3.1) due to neutron activation:

	<u>Without Shielding</u>	<u>With Shielding</u>
Heat Exchanger	9 R/h	2 mR/h
Pumps	300 R/h	50 mR/h

Without shielding, the above dose rates are sufficiently high that contact maintenance would be very difficult. Note that the unshielded dose rates are a factor of 90 to 3,000 times the assumed design criterion (100 mR/h). With shielding, the dose rates are very acceptable in comparison to the design criterion.

In the approximate calculations to estimate the above activation dose rates, the assumption was made that the material was type-304 stainless steel having the following composition:

<u>Element</u>	<u>Weight Percent</u>
Iron	67.7
Chromium	20.0
Nickel	10.0
Cobalt	0.025
Others	2.275 (Mn, Si, C, P and S)
Total: 100.000	

Note that the cobalt content was assumed to be only 0.025%, which may be as much as a factor of 4 lower than is sometimes found in type-304 stainless steel. This is significant since the Co-60 decay activity represents 80% (or more) of the indicated dose rates. Accordingly, it appears important to note carefully the cobalt content of stainless steel or nickel alloy components where neutron activation presents a radiation problem.

FIGURE 2.37 THERMAL NEUTRON FLUX VERSUS RADIAL DISTANCE FROM CENTER OF CORE

2.3 PRINCIPLES OF OPERATION

2.3.1 Preparation for Initial Startup

The first loading of the reactor and the first startup will differ somewhat from subsequent refuelings and startup. The core will be initially hand loaded with dummy fuel bundles before the primary heat transfer system is filled with sodium. The operators will check out the refueling and other mechanisms while the reactor and refueling cell are filled with air and are nonradioactive.

The subsequent steps are as follows:

1. Bring the initial loading of fuel into the new fuel storage tank and into the empty fuel decay tank.
2. Purge the refueling cell and reactor vessel with nitrogen and then with argon, and check for leaks.
3. With the shield plug in place, heat the primary sodium tank to 400° F by circulating hot argon.
4. Fill the primary sodium tank with sodium, start the pump pony motors, and operate the cold traps to establish the required sodium purity.
5. Exchange new fuel bundles for dummy fuel bundles in core.

The fuel bundles with control rods should be loaded first. This procedure is the same as that described for normal refueling, except that an entire core loading must be handled, requiring both the new fuel storage tank and the fuel decay tank to be used for transfer in and out. (The dummy fuel bundles can subsequently be used in the loading of another reactor).

2.3.2 Startup

After the refueling operation is completed, the reactor is ready for startup. The startup procedure is as follows:

1. Check the availability of the primary heat transfer system, and establish a primary sodium flow of 20% (by the pony motor drives).
2. Withdraw scram rods, one at a time. During this procedure, the reactor should not reach criticality. If it did reach criticality however this would be detected and the scram circuit would fire all of the scram rods that had been removed. In that event a change in the fuel loading must be made.
3. Withdraw shim rods until criticality is reached.
4. Raise the reactor power by adjusting the shim rods, and increase the primary sodium flow through the reactor until full operating power and temperature conditions are reached. The rate of power and temperature increase are limited primarily by safety of the reactor system, and the ability of the other systems to accommodate the change in temperature and power level.

2.3.3 Normal Operation

During normal operation, continuous measurement is made of neutron flux, flow and outlet temperatures of all fuel bundles. The shim control rods are operated in response to signals from the plant control system (Section 3.13).

Normally the shim rods are operated with as many rods fully withdrawn as is possible. One or two rods are at intermediate positions to provide fine reactivity

adjustments. The remaining rods are fully inserted.

2.3.4 Shutdown

The reactor is shut down on a regular annual basis for refueling, and at intermediate times if necessary to accommodate an emergency. Normal shutdown is accomplished by slowly inserting the shim control rods into the core to maintain the reactor slightly subcritical. This results in a slowly decaying power level. The rates of power and temperature change are limited to minimize thermal shock on components of the other systems. When the power level has reached a low value (about 5% of full power), the remainder of the shim rods are driven in and the scram rods are dropped in. The scram rods are dropped by stopping the vacuum pump which holds them in their fully withdrawn position.

2.3.5 Emergency Operation

Several types of malfunction can occur in the system and require the shutdown of the reactor. Only malfunctions which threaten to cause immediate core damage require the sudden insertion of the scram rods. A scram shutdown would be initiated by an indication of fuel channel flow blockage or by a sudden increase in neutron flux.

To minimize thermal shocks, reactor shutdown would be initiated by insertion of shim rods in all emergencies which do not immediately threaten core damage. For example, an emergency affecting the secondary sodium system would provide little incentive for a sudden scram because of the very large heat storage capacity in the primary sodium system.

2.4 HAZARDS AND SAFETY PRECAUTIONS

2.4.1 Hazards

The determination of when a malfunction changes from a maintenance problem to a hazard is difficult to make. For purposes of discussing hazards connected with the reactor system, the following classifications seem convenient:

1. Potential radioactivity hazards during normal operation,
2. Sodium fires,
3. Core damage (over and above normal percentages of failed fuel elements), and
4. Certain potential opportunities for mechanical damage, which might in some designs or circumstances be a hazard rather than a maintenance or cleanup problem.

The discussion of each of the hazards listed above is followed by a discussion of precautions which should be taken. In some instances the current design has not explicitly incorporated them, either because the drawings and analyses in a conceptual design effort were not carried to the degree of detail that would have been required, or because the derivation of what precautions were required had to await calculation of certain design characteristics, and was carried out near the end of the work. In either case, it is intended that if further work on the conceptual design were carried out, the precautions described below would all be included. It is not thought that any required features not now included would significantly alter the cost estimates.

2.4.1 Radioactivity During Normal Operation

2.4.1.1 Hazards

These arise from two sources; the direct neutron and gamma radiation from the core, and the radioactive material released from vented and failed fuel. The neutron and gamma radiation is a direct potential hazard to those working in the vicinity of the reactor system on maintenance or refueling. The neutron radiation can activate components and sodium, resulting in increased hazards during shut-down and maintenance, and increased radiation due to the activated sodium at locations far removed from the core. The emission of fission products from vented fuel, and of fission products and fuel from failed fuel, poses potential for increased radiation hazards at all points in the primary sodium system, but particularly during maintenance of primary system components. It also makes the cover gas radioactive, which is a potential hazard due to leakage paths, and because of the occasional need for entry into the refueling cell.

2.4.1.2 Precautions

Thick barriers of shielding materials reduce the amount of direct radiation to personnel who are working near the reactor system. Elaborate seals are provided at all penetrations through the containment barrier to prevent the escape of contaminated cover gas. Radiation monitors in the reactor building and periodic radiation surveys assure that the radiation exposure of personnel is kept at an acceptable level.

The procedures for maintaining the components of the primary heat transfer system include precautions against radiation due to activation of the component and due to leakage of contaminated gas and sodium. These procedures are de-

scribed in Section 3.1. Procedures for maintaining the components of the reactor system are described in Section 2.5.

2.4.2 Sodium Fires

2.4.2.1 Hazards

The possibility of sodium fires, with resultant damage to the plant and a release of caustic smoke causing hazards to the environs, exists for all sodium systems in the plant. For the reactor system, the problem is compounded by the fact that all sodium associated with the reactor system, including that in the fuel storage pools in the refueling cell, will be activated with Na-24 and Na-22, and will be contaminated with some concentration of fission products.

2.4.2.2 Precautions

Sodium fires are prevented in the reactor system by excluding air from contact with the system. The refueling cell is normally filled with argon gas, and the cover gas over the sodium of the primary heat transfer system is argon. The inert gas systems (Section 3.8) continuously monitor and control the purity of the argon. Material transfer locks are provided for bringing equipment into the cell with a minimum of air contamination.

2.4.3 Core Damage

2.4.3.1 Hazards

Any substantial damage to the core increases the release of fission products, aggravating the normal hazards involved in the radioactivity of the coolant and cover gas. If this damage extends beyond a very few fuel assemblies, it carries with it the possibility of significant nuclear energy releases, which might damage

the reactor structure, exposing the primary system and creating hazards of sodium fires and activity release. There also exists the non-nuclear problem of sodium vapor "explosions" due to rapid transfer of heat from fuel to sodium.

Damage to the core may arise from one of three immediate causes:

1. Accidental reactivity insertions,
2. Accidental flow reductions, and
3. Local damage due to individual assembly flow blockages or fuel failures.

2.4.3.2 Precautions

The precautions taken against core damage consist of the core and primary system instrumentation, the control rods and drive systems, the automatic control logic circuits and the control room annunciators, various design measures to limit the magnitude and/or rate of possible accidental reactivity insertions and various design measures to limit the magnitude and/or rate of possible accidental flow reductions. In addition, the reactor vessel and shield plug are probably capable of withstanding a significant explosive energy release, sufficient in any case for sodium vapor explosions involving only a few assemblies.

This set of precautions is intended to prevent, with very high reliability, any core damage extending beyond seven fuel assemblies, and in most cases beyond one fuel assembly.

The core instrumentation will consist of neutron monitors covering the range from startup to full power, with essentially instantaneous response; flow measurement on each

fuel assembly with a time constant for response of 500 msec or less; and outlet temperature measurement on each fuel assembly with a time constant of 800 msec or less. The primary system instrumentation will include thermocouples above the core to measure mixed mean outlet temperatures; six or more area monitors for fission products, located above the core to take sodium samples from specified sectors of the core outlet sodium; activity monitors in the cover gas; inlet pressure sensors in the core inlet plenum; and a limited number of acoustic detectors to detect sodium voiding.

Scram will be initiated if flux increases by 15%, if core flow decreases by 10%, if core outlet temperatures increase by 10%, if inlet plenum pressure decreases unexpectedly, and if local boiling is detected. In addition, abnormal flow or temperature indications on an individual assembly will initiate scram. A combination of flow and temperature will initiate scram with a smaller signal than would be required from either separately. In addition, indications of abnormal conditions from adjacent assemblies will be given increased weight. The detailed logic of the automatic scram based on local flow and temperature measurements has not been specified.

The reliability, and therefore the redundancy, requirements for the instrumentation have not been determined. On the basis of studies done for the FFTF, however, it is recommended that the assembly outlet temperature thermocouples should have four sensors, with one being a spare, and the measurement based on two-out-of-three logic.

The delay from receipt of a scram indication to initiation of rod movement is 150 milliseconds. Rod acceleration is 3 g's. To ensure scram when needed, there is a

100% redundant set of backup rods, with independent drives and control circuits. (It should be noted that reliability analyses may eventually demonstrate that a backup system is not needed.) The scram and backup rods will have sufficient reactivity to compensate for up to seven fuel assemblies voiding and compacting, sticking of a rod, and any other sources of reactivity addition which might be triggered by accident conditions.

With these instrumentation and control provisions, analyses reported in the Task III report demonstrate that slightly over one dollar of reactivity may be tolerated in a step insertion. With slower insertions, larger amounts can be handled, possibly up to a \$4.25/second ramp. All sources of accidental reactivity insertion must be analyzed and held below these limits. Such sources include:

1. Rod runout,
2. Rod blowout, if shown to be possible,
3. Radial core movement,
4. Axial fuel movement within the cladding, and
5. Sodium voiding of an assembly.

This is not necessarily an exhaustive list. It is not anticipated that the provisions necessary to limit these sources of reactivity, such as shim rod worth limits and radial core clamping, will constitute serious compromise in design.

Analyses of flow reductions in the Task III report indicate that up to 60% step flow reduction can be tolerated. If the flow reduction takes 5 seconds, it can be as severe as 90% without causing damage. In the absence of reliability

figures for piping and structural members, the vessel, inlet plenum, and piping will be designed so that no single structural failure results in a flow reduction that causes fuel damage. In addition, pump inertia and emergency power will ensure that loss of station power does not cause an unsafe condition. Multiple loops ensure that sudden seizure of one pump, or loss of a heat exchanger, still allow adequate cooling to take place. Fuel assembly inlets will be designed to prevent transport of large bubbles from one assembly inlet to an adjacent one, and to prevent any reduction of greater than 50% in assembly flow due to an inlet blockage. Fuel pins are vented to avoid internal pressure and minimize the fuel and cladding movement involved in any pin failure. Flow, temperature, and boiling detectors should be adequate to limit any local damage propagation to a maximum of seven assemblies.

Finally, the large reservoir of sodium in the primary tank provides assurance that the core cannot be uncovered by pipe breaks, and that a large heat sink is available for temporary removal of decay heat (for over an hour). At ~ 6% of full power natural convection cooling is sufficient to prevent fuel damage.

2.4.4 Mechanical Damage

2.4.4.1 Hazards

Examples of mechanical malfunctions or damage which could directly result in a hazard in some designs or circumstances would be dropping the shield plug on the core, dropping a fuel assembly during refueling, failure of the refueling machine while in the process of transferring an element, breakage of the refueling cell windows during refueling, and earthquakes.

2.4.4.2 Precautions

The design of the shield plug and its lifting mechanism is such that it will not reach the core, should it fall while being lifted. Dropping a fuel assembly within the refueling cell may pose a severe maintenance and repair problem, but should not cause any hazard to personnel, due to the protection afforded by the refueling cell. Failure of the refueling machine causes no hazards, since the fuel assembly can be cooled by natural convection without overheating. Protection must be furnished to ensure that accidental breakage of the refueling cell window either cannot occur, or will not result in release of significant activity. The plant will be designed to withstand earthquakes up to 0.2 g horizontal simultaneous with 0.15 g vertical.

2.4.5 Routine Operational Precautions

In addition to the design precautions noted above, numerous routine operational precautions should be observed. These include:

1. Exercise of control rods and periodic determination of worth,
2. Measurement of power coefficients on a periodic basis,
3. Calibration of critical instruments,
4. Removal and replacement of control rods prior to excessive neutron damage,
5. Monitoring and sampling gases in refueling cell for excess radioactivity prior to personnel entrance,

6. Check of seals in refueling cell decay storage plug and reactor plug,
7. Routine check of electronic safety circuits with built-in scram simulating checks,
8. Measurement of total worth of fuel on a periodic basis, and
9. Calibration of xenon decay curves and recheck against these curves on a regular basis.

2.5 MAINTENANCE

2.5.1 Refueling Equipment

The fuel shuffling machine and the fuel transfer machine, as well as other equipment directly associated with the refueling operation, will be serviced while the reactor is in operation. At the conclusion of the refueling operation, and after the reactor has resumed operation, excess sodium which has dripped on to the floor of the refueling cell will be cleaned up, and temporary floor coverings will be removed. These operations are handled with the overhead manipulator.

After all the used fuel has been shipped out of the refueling cell, the cell and equipment will be given a final cleanup. Then, operation of the mechanisms will be checked out. Any indication of faulty operation will call for suitable repairs to be made. When practical, minor repairs or adjustments will be made in the refueling cell by remote manipulator equipment. When this is not feasible, a man will enter the refueling cell with protective clothing and breathing apparatus to make the repairs. All major pieces of equipment are designed in such a way that the operating parts can be removed from the refueling cell for major repairs or replacement. The fuel shuffling machine can be

disassembled into three parts, each of which can be removed from the refueling cell through a two-foot diameter air lock.

2.5.2 Reactor Thermocouples

The reactor outlet thermocouples are usually tested during normal operation, prior to refueling. During the shutdown for refueling, those instrumentation capsules which were found to contain defective thermocouples are replaced. Replacement would be required if an inconsistency was detected in the readings from an individual group of three thermocouples. A fourth thermocouple in each group can be activated during reactor operation in the event of failure of any one of the others, and immediate replacement of the capsule will not be required. In the event that only one thermocouple in any group remains operable, the reactor must be shut down for replacement of the capsule. Replacement of the thermocouple capsule is done by using the overhead manipulator and requires unplugging the electrical leads, unclamping the capsule, and lifting it out vertically. A new unit would be installed by lowering it into place, clamping, and making the electrical connections.

Neutron monitors are installed through the shield plug in a manner similar to the outlet temperature thermocouples.

2.5.3 Control Rod Drives

Control rod drives are replaced either on a routine basis or upon indication of a malfunction during operation. Installation and removal are done in the same manner as the thermocouple capsules. Repairs can not be made with the drive in place. As with the thermocouples, an indication of faulty operation would be determined before shutdown, and a replacement unit would be brought into the refueling cell.

2.5.4 Inlet Flow Meters

As a routine operation, the gas pressure in the flow-measuring tanks would be fluctuated over a narrow range. Readings from each flow meter will respond to this fluctuation unless its sensing tube is plugged. In the latter case, the bank of tubes can be purged with sodium by lowering the gas pressure in the flow measuring tank until sodium flows out of each tube. This will tend to free any blockage.

The electrical sensors can be repaired by removing the entire probe unit.

2.5.5 Shield Plug

The lifting mechanism for the shield plug is accessible for maintenance when the shield plug is in the down position. If necessary, the two vertical lead screws can be withdrawn through the top of the refueling cell after disconnecting the drive support arms by unbolting. The main seal of the shield plug (the design is not shown in the drawings) would be designed in such a way that it can be replaced with the shield plug in the down or seated position. This seal should be replaced after each refueling.

Major repairs on the lower end of the shield plug are not feasible and replacement of the entire shield plug would require that a special hole be made in the refueling cell wall or ceiling.

2.5.6 Reactor Core Structure

The core structure unit, which contains the bottom grid plate and the structure surrounding the core, is assembled into the reactor vessel as a unit, and is bolted through the bottoms of the 24 temporary fuel storage positions around the sides of the core. Removal of this unit would be difficult, and would require the removal of the fixed intermediate plug

surrounding the shield plug. Removal and replacement of both of these large components would require that a large hole be made in the wall of the refueling cell.

The fuel bundle supports, which are

individually bolted to the bottom grid plate, are removable without removing the entire core structure. Special tools would be required for this operation. Note that the bolts are staked in place, and require a definite minimum torque.

4

SECTION III

PLANT SYSTEMS

3.1 PRIMARY HEAT TRANSFER SYSTEM

3.1.1 Introduction

3.1.1.1 System Function

The function of the primary heat transfer system is to transfer heat from the reactor core to the secondary heat transfer system.

3.1.1.2 Summary Description

The primary heat transfer system components and the reactor core are located entirely within a large tank which is filled with molten sodium. The sodium within the tank is made radioactive by its contact with the reactor core neutron flux. The components of the primary heat transfer system are submerged in the sodium and are accessible through removable shielding plugs. The following are the major components of the primary heat transfer system:

1. Primary sodium tank,
2. Containment vessel,
3. Reactor vessel,
4. Intermediate heat exchanger, (IHX) and,
5. Primary sodium pumps

Sodium is forced upward through the reactor core by three centrifugal pumps which act in parallel. The sodium is heated as it passes through the reactor core. At the reactor outlet, the sodium mixes and flows through three parallel pipes to the intermediate heat exchangers. The sodium is cooled as it passes through the tubes of the intermediate heat exchangers. The cooled

sodium is discharged within the tank, where it mixes and flows to the suction side of the three centrifugal pumps.

The decision to have primary sodium flow on the tube side of the IHX meets the objectives of this study. Primary sodium flow on the shell side may eventually prove to be more attractive. A discussion of this alternative is presented in the Task III report.

Flow through the tubes of any one intermediate heat exchanger can be stopped with the reactor continuing to operate, at reduced power. This will permit maintenance of the components (except the IHX) of the one secondary heat transfer loop which is associated with the given intermediate heat exchanger. Flow through the tubes is stopped by gradually increasing the pressure of the cover gas over the selected intermediate heat exchanger. When this pressure equals or exceeds that of the reactor vessel cover gas, flow to the unit will stop and the level of sodium in the tubes will be depressed by an amount corresponding to the pressure drop through the still active units.

3.1.1.3 Design Requirements

The primary heat transfer system is required to utilize liquid sodium to transport heat from the reactor core under all normal and emergency conditions. The following are specific requirements of the primary heat transfer system:

1. The flow rate of sodium must be smoothly variable over the range of normal operation to allow maintaining the reactor inlet and outlet temperatures within 20° F of the nominal full-power values.

2. Forced circulation of sodium must be assured under all emergency conditions.
3. No throttling valves or tight shut-off valves are to be used to control the sodium flow.
4. All components, which are not designed for the life of the plant, must be accessible for maintenance and repair.
5. All penetrations through the containment vessel must be adequately sealed to prevent both the leakage of the reactor building atmosphere into the system and the leakage of cover gas and sodium from the system.
6. The pressure of sodium on the secondary side of the IHX must be greater than the pressure in the primary heat transfer system, to prevent leakage of the primary sodium through a defective IHX.
7. No means should be provided for inadvertently lowering the sodium level below the top of the core.
8. The reliability of the components must be such that the plant unavailability is less than 10%:
 - a. Unavailability due to component failure must be less than 5%, and
 - b. Unavailability due to instrumentation failure must be less than 5%.
9. The temperatures and the flow rate of the primary sodium are to be determined by optimizing all of the plant design parameters.
10. The total fast neutron ($E > 1.0 \text{ MeV}$) dose to any component must not exceed $10^{21} \text{ neutrons/cm}^2$.

3.1.2 Detailed Description

3.1.2.1 Primary Sodium Tank (J.H. Germer)

The entire primary sodium inventory, with the exception of that in the primary cold trap system, is contained in a single primary sodium tank which is 48 feet in diameter and 40 feet deep, as shown in Figures 3.1, 3.2, and 3.3. This vessel is constructed of 1 inch thick stainless steel. It is not sealed at the top. Its inert cover gas communicates with the gas in the space between the primary sodium tank and the primary containment vessel. The primary containment vessel is insulated on the inside and cooled on the outside with nitrogen. This vessel is maintained near room temperature and is not required to withstand deformation resulting from thermal expansion. This arrangement permits the various penetrations through the biological shield for the reactor, heat exchangers, and pumps to be made without the need of complicated flexible joints. Each of these large components is separately suspended from the upper shield structure.

Loose fitting baffles are located at the top of the primary sodium tank, with sliding joints provided for differential thermal expansion. In order to prevent the establishment of thermal convection loops past these baffles and migration of sodium vapor to the cold surfaces, an argon circulating system is provided. Argon is removed above the surface of the sodium and at a temperature essentially equal to that of the sodium. It is then cooled in such a way that the majority of the sodium vapor is condensed and drained. Periodic defrosting of the argon cooler is done to remove sodium which is frozen on its surfaces. The cooled argon is then returned to the space between the primary sodium tank and the primary containment vessel. The argon gas then passes downward through the openings between the baffles at a velocity sufficient to prevent upward natural convection of

sodium-laden argon. This argon system is discussed in detail in Section 3. 8. 2. 2.

a. Primary Sodium Tank Support

The weight of the primary sodium tank is supported from its upper flange by 48 tension bolts which are 2 inches in diameter and which extend into the upper shield structure. They are freely mounted to permit approximately 2-1/2 inches of radial thermal expansion of the primary sodium tank. The bolts are located in the atmosphere of the reactor argon, and their mounting is sealed at the upper end.

The upper flange of the tank is a reinforcement which maintains the circular shape of the upper edge of the tank, and which provides a location to anchor the supporting tension bolts. This flange is located above the normal sodium level of the tank.

Horizontal alignment of the primary sodium tank is provided by a series of shear lugs which connect the side of the tank through the primary containment vessel to the concrete shield structure. These shear lugs permit free radial expansion of the vessel, but resist in shear any horizontal seismic displacement which might occur.

b. Insulation and Cooling

The primary containment vessel (Figure 3.2) is insulated on its entire inner surface with a layer of insulation which is six inches thick. The insulation consists of reflective stainless steel in the form of many thin sheets separated by argon gaps. This is made up in preformed panels that are suspended from the vessel. A gap of approximately one foot separates this insulation from the primary sodium tank. Inert insulation is required to prevent impurities from forming in the argon cover gas system.

A carbon steel liner is provided for the concrete shielding. The space between the vessel and the liner is cooled by circulation of nitrogen gas to remove heat which passes through the insulation. The nitrogen also carries away the smaller amount of heat that is formed by gamma rays and neutrons in the shield near the reactor core.

c. Baffle Plates

The baffles at the top of the tank are plates of carbon steel suspended by tension bolts from the upper shield structure, including the lower surfaces of the various shield plugs. These plates are shown in Figure 3.2 and in Figure 2.7. The plates are constructed in separate pieces of approximately 16 square feet each, and they are overlapped in a manner that minimizes downward flow of argon. Edges of these plates are joined to the various shield plug shrouds with a sliding joint wherever possible, also to minimize this argon flow.

3.1.2.2 Reactor Vessel (J.H. Germer)

The reactor vessel is located within the primary sodium tank and is used to contain and support the reactor. It provides flow isolation for the reactor inlet plenum and for the reactor outlet plenum. It is shown most clearly in Figure 2.7 and also in Figure 3.2. It is supported from an upper flange which bears on the upper shield structure.

The lower half of the reactor vessel is exposed to reactor inlet temperature on both the inside and the outside and therefore is not subjected to thermal stress. The upper section is exposed to the same inlet temperature on the outer surface, but the inner surface contains sodium at reactor outlet temperature which is 340° F hotter. Since this large temperature difference would overstress a single wall of stainless steel, the reactor vessel is constructed in two layers which are separated

FIGURE 3.1 REACTOR AND PRIMARY SODIUM TANK
ARRANGEMENT-PLAN VIEW ABOVE
GRADE (SIMILAR TO GE DRAWING
174F111 SH. 1)

FIGURE 3.2 REACTOR AND PRIMARY SODIUM TANK
ARRANGEMENT—ELEVATION (SIMILAR
TO GE DRAWING 174F111 SH. 2)

FIGURE 3.3 REACTOR AND PRIMARY SODIUM TANK
ARRANGEMENT-PLAN VIEW BELOW
GRADE (SIMILAR TO GE DRAWING
174F111 SH. 3)

by a gap containing argon gas. The outer layer is not subjected to thermal shock, since the inlet temperature of the reactor remains essentially constant. The upper end of the outer layer, however, sustains a steady-state temperature gradient from the inlet temperature of 810°F at the level of the sodium surface to approximately 100°F at the top support flange.

The inner layer of the vessel, or the liner, normally operates at reactor outlet temperature of 1150°F. This temperature can rapidly (in a few seconds) decrease to a value close to the inlet temperature. However, the upper edge of the liner is not restrained, and the liner is not required to serve any structural purpose other than containing the sodium.

The two layers of the reactor vessel join concentric pipes to the three intermediate heat exchangers. The argon insulating gap is maintained throughout, where outlet temperature sodium is contained. The detailed design of the expansion joint in the concentric pipes is discussed in the IHX design description. The decision to use argon gas insulation in the primary heat transfer system components meets the objectives of this study. Another approach may eventually prove to be more attractive. A discussion of steel and stagnant sodium insulation is presented in the Task III report.

The space between the inner and outer layers of the reactor vessel is vented near the lower end to the main region of the primary sodium tank. Since the argon cover gas over the reactor during operating flow is at a pressure of about 5 psi above that of the rest of the tank, the level of sodium in the insulating gap will remain at a level of about 15 feet below the level of the sodium in the primary sodium tank. When reactor flow is stopped, this annulus will fill with sodium.

At a level above the inlet plenum, the reactor vessel is provided with a flange onto

which the reactor structure is bolted. The bolts withstand the upward force upon the reactor structure.

The inlet plenum forms the lower end of the reactor vessel, and is supplied with sodium from three sodium pumps. Two pipes are provided from each pump in order to minimize the back flow which might occur if a single pipe were to fail. The inlet plenum is surrounded by an outer shell which normally is not pressurized, but which will prevent gross core bypass in the event of failure of the inlet plenum. This outer shell is not joined to the pipes, but the clearance is sufficiently small to prevent excessive leakage during an accident.

3.1.2.3 Intermediate Heat Exchanger (J.H. Germer)

The intermediate heat exchanger (IHX) transfers heat from a primary (and highly radioactive) circuit of reactor coolant sodium to a secondary loop. Three units are operated in parallel primary circuits, but are connected to three independent secondary loops. Each IHX is removable as a unit from the primary sodium tank along with a shield plug, and may be repaired or replaced.

The following are the basic requirements for the intermediate heat exchanger design:

1. The pressure drop on the primary side should be low. This considerably simplifies the control of the sodium level, both above the IHX and above the reactor outlet.
2. The ends of the individual tubes of the IHX should be accessible for inspection and for plug-welding in the event of a failure.
3. There should be a minimum tendency for sodium to cling to the primary side of the IHX if it is withdrawn.

4. The IHX should be resistant to damage by thermal shock, which would occur if the reactor power were suddenly scrambled.
5. The flow distribution in the IHX should be stable under both full and part load conditions. This requires that the primary sodium stream must flow downward and that the secondary stream must flow upward in the region where heat transfer occurs.
6. The region above the floor level must be adequately shielded against radiation from the primary sodium as well as against direct radiation from the reactor. The extremely large secondary sodium pipes that are required essentially preclude the possibility of applying effective shielding except above the pipe transition section, where removable shield blocks are used.
7. Flow distribution should be uniform and should approach idealized counter-flow in order to assure efficient operation.
8. The IHX must withstand a higher pressure on the secondary sodium side of about 100 psi during normal operation, with a possibility of 250 psi during an accident condition. The secondary pressure is always maintained at a higher level than the primary pressure in order that any leakage will not cause radioactive primary sodium to leak into the secondary system.

Figure 3.4 shows the design of the IHX. The dimensions on the figure are superseded by those in Table 3.1. The main body of the heat exchanger is contained in a cylindrical section which is 9-1/2 feet in diameter and 20 feet long and which has spherically-disked tube sheets at each end. Primary sodium

enters the upper tube sheet and passes downward within 5472 tubes to exit through the lower tube sheet.

Secondary sodium enters through a pipe which extends to near the bottom of the heat exchanger. It flows upward past the outside of the tubes, and leaves through an annular pipe region which surrounds the inlet pipe. A transition section above the floor level separates the inlet and outlet secondary sodium flow streams from the concentric pipes into two separate pipes.

A large (40-inch) pipe carries flow from the exit plenum of the reactor to the inlet of the IHX. A flow distribution header surrounds the IHX and distributes the flow from the pipe to the top tube sheet of the IHX. Surface turbulence of the sodium is minimized by a flow baffle below the liquid level. A close fitting joint between the lower end of the heat exchanger and the surrounding structure minimizes primary flow leakage past the heat exchanger.

Table 3.1 shows the dimensional and operational parameters of the intermediate heat exchanger. A detailed discussion of the design features follows.

1. Tube Sheets

The upper and lower tube sheets are spherical segments in order to withstand the internal pressure of 250 psi without excessive stress. The tubes join the spherical surface in a radial direction. Welding projections are machined as part of the tube sheets.

2. Outer Shell

The shell is 9-1/2 feet in outside diameter, 1-3/8 inches thick, and 20 feet long. It withstands the 250 psi internal pressure under accident conditions. Its lower end forms a close fit with the fixed outer structure to minimize bypass leakage of primary flow.

FIGURE 3.4 INTERMEDIATE HEAT EXCHANGER
(SIMILAR TO GE DRAWING 27E229)

TABLE 3.1

INTERMEDIATE HEAT EXCHANGER DATA

Shell-Side Fluid	Secondary Sodium
Tube-Side Fluid	Primary Sodium
<u>Physical Description (Per Unit)</u>	
Heat Transfer Area, (based on tube o.d.)	25,800 ft ²
Number of Tubes	5472
Outside Diameter of Tubes	0.900 inches
Thickness of Tubes	0.071 inches
Tube Length, Approx.	20 ft.
Outside Diameter of Shell	9.5 ft.
Material	Stainless Steel
<u>Operating Conditions, Normal Full Power (Per Unit)</u>	
Primary Sodium Flow Rate	26.87×10^6 lb/hr
Secondary Sodium Flow Rate	21.42×10^6 lb/hr
Primary Sodium Inlet Temperature	1150° F
Primary Sodium Outlet Temperature	808.8° F
Secondary Sodium Inlet Temperature	646.1° F
Secondary Sodium Outlet Temperature	1071.4° F
Primary (Tube Side) Pressure Drop	3.0 psi
Secondary (Shell Side) Pressure Drop	14.8 psi
Primary Sodium Velocity	8.7 ft/sec
Design Pressure (Shell Side)	250 psig
Corrosion Allowance (Tubes)	0.030 inches

Four partial baffles are attached to the inside of the shell, and the ends of the shell are welded to the two tube sheets.

3. Support of IHX

The weight of the IHX is taken by a thermal sleeve which joins the outside of the outlet pipe with the shield plug. This thermal sleeve also serves as a seal against cover gas leakage out into the reactor building.

4. Secondary Piping

The inlet pipe passes downward through the inside of the outlet pipe, and extends to near the lower tube sheet of the heat exchanger. At that point, flow baffles direct the flow outward across the tubes. The inlet and outlet streams are separated by a double-wall tube which forms an argon gas-filled annulus. The annulus minimizes heat transfer between the two streams. The lower end of this annulus

is open, and the sodium level in the annulus is controlled at a level near the bottom by regulating the argon pressure in response to a level sensor located in the annulus. In order to minimize the flow resistance at the upper transition between the inlet and outlet pipes, the inner (inlet) pipe is set at a slight angle from vertical.

Construction of this concentric piping is arranged in such a way that the inner pipes can be inserted into the heat exchanger before making a final weld closure in the outer pipe immediately above the top tube sheet.

A separate outer pipe surrounds the inlet pipe in the body of the heat exchanger, and serves primarily as a centering guide for the inlet pipe and as a support for the tube baffles. This pipe is supported from the bottom tube sheet and is perforated with large access holes (not shown in the drawing) to assist in assembly of the heat exchanger tubes.

5. Heat Exchanger Tubes

The 5472 tubes are arranged in an involute pattern of 114 involutes with 48 tubes each. Individual tubes are 0.900 inches outside diameter and 0.071 inches thick. The tubes are straight except for a "sine wave" bend to allow for thermal expansion and a bend near each end to allow the tube to be welded perpendicular to the surface of the spherical tube sheets. Welds to the tube sheets are internal butt welds to short machined tube stubs on the tube sheets. The tubes are designed to withstand an external pressure of 250 psig and have a corrosion allowance of 0.030 inches.

Tubes are spaced by two sets of flow baffles. One set of baffles extends outward from the inner pipe, and the other set extend inward from the outer shell. Each baffle supports half of the tubes only. These baffles are perforated in such a way that a

large portion of the flow passes directly through the baffle, an arrangement which offers a relatively low pressure drop as well as some interchannel flow mixing. The "sine wave" bends in the tubes are made in a circumferential direction, with all bends in phase between two baffles.

6. IHX Support Structure

The intermediate heat exchanger as shown in Figure 3.4 is surrounded by a permanent structure which guides the radial position of the heat exchanger and forms the flow channel connecting the outlet plenum of the reactor with the inlet of the heat exchanger. Since the temperature of the sodium entering the heat exchanger is at a much higher temperature than the surrounding sodium, the structure is made in two layers, separated by an argon gas space in order to minimize thermal stresses.

The outer layer of the two-layer shroud is supported directly from the upper shield structure. It is exposed to a temperature gradient from 810° F at the surface of the sodium to about 100° F at the point of support. The inner shroud layer is jointed to the outer one near the lower end of the heat exchanger, and is free at the upper end between the sodium level and the bottom of the shield plug.

The large pipe extending between the reactor outlet plenum and the intermediate heat exchanger carries sodium at a temperature 340° F higher than the surrounding sodium. This requires the use of a separate liner pipe which is free to expand independently to avoid severe thermal stress. Both the liner pipe and the outer pipe must accommodate an axial expansion of about 2.5 inches as the reactor vessel is heated from its installed temperature, and a contraction of about 1.5 inches when the primary sodium tank is cooled to refueling

temperature (400° F). An additional expansion of 1.2 inches must be accommodated by the liner pipe when the reactor is in operation.

Other requirements of the expansion joint are that it be designed to accommodate any leakage that would occur, both of sodium and of argon gas. Figure 3.4 shows the details of the expansion joint. The outer pipe accommodates expansion by slip joints at the ends of an intermediate pipe, where the slightly enlarged ends of the intermediate pipe slide into recesses in the ends of the adjoining pipe. Leakage through this joint will cause argon bubbles to leak out into the primary sodium tank, and this can be accommodated if the argon circulating system is adequate. The use of a slip joint meets the objectives of this study. The use of another means to accommodate the expansion may eventually prove to be more attractive. A discussion of other possibilities is presented in the Task III report.

The liner pipe has a similar expansion joint, and in this case leakage will be outward due to a slight hydrostatic pressure differential. Since leakage of high temperature sodium would cause serious thermal stress on the outer pipe, an intermediate leakage shroud catches this leakage and drains it into a sump. This leakage shroud is a loose piece which acts only as a catcher for the leakage flow.

Another short length of pipe forms a gas seal to accommodate the approximately 2-psi difference in argon pressure at the two ends of the annulus resulting from the liquid pressure drop through the tube. This pipe is a close fit at one end to the inside surface of the outer pipe, and a close fit at the other end to the outside surface of the inner pipe. Again, any leakage past the seal must be accommodated by the argon circulating system.

3.1.2.4 Primary Sodium Pumps and Drives (C. E. Boardman)

Three parallel primary coolant pumps are arranged within the primary tank in such a manner that the drive motors and seals are located outside the refueling cell in the air zone. (See Figure 3.2). Each of the primary coolant pumps is a centrifugal-type, free-surface, single suction pump with a removable inner barrel or casing. These units are designed to utilize a sodium-lubricated self-aligning hydrostatic bearing, gas lubricated face-type shaft seals, and a vertical wound rotor drive motor with a liquid rheostat speed control. The choice of a wound rotor motor meets the objectives of this study. Another choice may eventually prove to be more attractive. A discussion of alternatives is presented in the Task III report.

Each pump is supported within the reactor top shield deck and extends vertically downward inside a perforated concentric support cylinder. Its downward flow discharge nozzle enters a labyrinth sealed discharge plenum. The pump check valve is contained within the discharge nozzle and is thus made accessible for repair work by removing the pump.

The discharge plenum is an integral part of the fixed pump support cylinder into which the pump may be inserted and withdrawn. The plenum has a sliding fit pressure balanced connection with the discharge piping. The connection greatly reduces piping reaction and differential expansion problems. The two 30-inch diameter pipes which connect each pump discharge plenum with the reactor vessel inlet plenum are rigidly attached to the reactor vessel but connect and pass through the pump discharge plenum by way of a pair of labyrinth seals. The pipe inlet flow passes radially into the pipes between the seals, (see Figure 3.2) thus,

TABLE 3.2

PRIMARY SODIUM PUMP CHARACTERISTICS

Type	Centrifugal, shaft sealed, sump type
Fluid	Radioactive sodium
Material	Stainless steel
Number	3
Flow Rate, lb/hr	26.873×10^6
Flow Rate, gpm	63,800
Pump Speed, rpm	670
Total developed head, ft of Na	237
Hydraulic Horsepower, hp	3692
Net positive Suction Head, ft of Na	51
Operating Temperature, ° F	809
Design Temperature, ° F	900
Cover Gas Pressure, psia	15
Depth of Impeller, ft	10
Shaft length (from coupling to pump impeller), ft	23
Withdrawal length, ft	32
<u>Main Pump Drive Characteristics:</u>	
Drive Type	Vertical, Wound-Rotor Induction Motor
Speed Control	Liquid Rheostat in Rotor Circuit
Number	3
Synchronous Speed, rpm	720
Power Rating, hp	4000
Electrical Supply	3 phase, 60 cps, 4000 volts
<u>Auxiliary Pump Drive Characteristics:</u>	
Drive Type	Squirrel Cage Pony Motor
Number	3
Pump Speed, rpm	134
Power Rating, hp	30

achieving a sliding pressure balanced connection.

The pump shaft length from the top coupling to the pump impeller is 23 feet. The overall withdrawal length down to the end of the discharge nozzle is 32 feet. During

normal three-loop operation, each pump will operate at 63,800 gpm net flow with a total developed head of 237 feet. Performance and design information are presented in Table 3.2.

The pump shaft is supported by an oil lubricated thrust bearing located within the

sealing unit above the reactor operating floor. Shaft alignment is maintained by a self-aligning hydrostatic radial bearing located in the sodium pool. High pressure sodium from the pump discharge is supplied to the bearing to maintain impeller concentricity. The bearing overflow is allowed to leak back into the reactor tank through small holes in the pump barrel.

An oil seal unit provides lubrication and cooling for the pump thrust bearing and mechanical seals. The seals are gas-tight and isolate the reactor atmosphere from the reactor building air zone.

Each primary pump unit is driven by a vertical 4000-hp, 720-rpm (synchronous speed), three-phase, wound rotor motor. Each motor has a liquid rheostat to provide speed control from 20% to 100% of rated pump speed.

Normal inspection and maintenance of the pump seals, the motor, and the coupling is accomplished from the operating floor during reactor operation. After an extended period of time it may be necessary to remove the pump assembly for maintenance. The pump housing contains the pump shaft, the radial hydrostatic bearing, the impeller and the outlet check valve. This housing is designed for convenient removal as a unit from the primary tank without disturbing the connecting primary piping. This is done in a manner similar to that of the Fermi reactor, except that the pump discharge nozzle plugs into a discharge plenum instead of into the outlet piping. Replacement is accomplished by raising the pump unit from the primary tank into an air-tight container. The container and contents are then transported by the reactor building crane to the maintenance and decontamination cell in a manner similar to that followed when removing the intermediate heat exchangers.

3.1.3 Principles of Operation

The primary heat transfer system is filled with sodium by the sodium service system which is described in Section 3.5. Prior to filling, the components of the system are heated to 350° F by hot gas from the argon system, which is described in Section 3.8. The argon system also assures that the temperature of the filled primary system never drops below 350° F.

When the reactor system is not in operation, sodium is circulated isothermally through the primary heat transfer system. The primary sodium pumps are driven at 20% of full speed by the low-power pony motors. Any decay heat from the reactor system is transported to the secondary heat transfer system, to maintain the sodium temperature below 400° F.

When the reactor is being brought up to full power, the flow rate of sodium through the primary heat transfer system is smoothly increased by the variable speed, wound-rotor motor, main drives of the sodium pumps. A description of the plant start-up procedure is presented in Section 3.13.

Over the range of normal operation of the plant, the temperatures of sodium entering the reactor core and of sodium leaving the reactor core are held within 20° F of the nominal full power values. This is done by varying the speed of the primary sodium pumps as the reactor power varies. A description of plant control over the range of normal operation is presented in Section 3.13.

During an orderly shutdown, the speed of the primary sodium pumps is decreased smoothly to 20% of full speed. At that speed the pony motors are energized and the main drives are de-energized.

When a reactor scram occurs, the pony motors are immediately energized. The main drives are de-energized at the discretion of the plant operators. The primary sodium pumps then coast down to 20% of full speed and are maintained at that speed by the pony motors.

Mass transfer from the surface of the reactor fuel pins and other hot surfaces might cause the formation of a corrosion product scale on the colder regions of the primary sodium side of the IHX tubes. Periodic removal and cleaning of each IHX might be necessary in order to remove this scale. If left in place, the scale could result in poorer heat transfer and higher primary-side pressure drop.

3.1.4 Safety Precautions

The following are measures which must be taken to assure the safe operation of the primary heat transfer system:

1. The atmosphere of the reactor building must be continuously monitored for radioactive contaminants.
2. The seals at each penetration through the containment vessel must be regularly examined, and repaired or replaced when necessary.
3. The thermocouples on the surface of the containment vessel must be regularly monitored to assure proper distribution of the cooling gas.
4. The thermocouples on the surface of the primary sodium tank must be regularly monitored to assure proper distribution of the primary sodium flow.
5. Materials samples which are located in the system must be periodically removed and examined to evaluate the following effects:
 - a. Mass transfer (including intergranular corrosion).
 - b. Irradiation damage to the components of the primary heat transfer system.
 - c. Activation of the components of the primary heat transfer system.
6. Prior to removing either the IHX or the sodium pump, the seals of the transfer cask must be carefully checked. The cask must be properly purged of oxygen. It must be carefully drained of secondary sodium during lifting.
7. The sodium pumps must be regularly monitored for vibrations and evidence of cavitation.
8. The bulk sodium temperature must be monitored during load changes to detect abnormal values.
9. Sodium samples must be taken regularly, especially from the reactor outlet plenum and the cold traps, and examined for either delayed neutrons or a change in the normal fission product spectrum.
10. The sodium must be continuously monitored for high oxygen level.
11. The sodium must be periodically sampled and examined to determine the complete spectrum of impurities. Careful attention should be paid to carbon level as an indication of possible oil in leakage.
12. During shutdown the check valves should be exercised to confirm their operability.
13. The primary sodium pressure drop through the IHX's must be periodically measured to detect the build-up of scale.

14. Local strain gages on components of the primary heat transfer system must be periodically monitored for unusual indications.
15. Both the argon and the nitrogen regions of the primary heat transfer system must be periodically monitored for leakage through the containment vessel.

3.1.5 Maintenance (J. H. Germer)

Wherever possible, provisions have been made for maintenance of the components of the primary heat transfer system. The primary sodium pumps and intermediate heat exchangers, the major components, are each constructed as part of a shield plug which can be removed in the upward direction in order to make repairs or replacement. Removal is made into a container which is carefully designed to avoid contamination from the highly radioactive primary sodium system, and to prevent air from entering the system.

Before a replacement is made, the space below the shield plug and above the component to be removed is thoroughly purged with fresh argon to prevent the release to the building of highly radioactive xenon or krypton. The technique is similar to that performed on the reactor cover gas before refueling. Note that only the space above the particular component need be purged, since the remainder of the cover gas is isolated by the component shroud.

Removal of a primary sodium pump would be preceded by the removal of the drive motor. A gate valve is then installed at the top of the shield floor, and a container with another gate valve is installed above it. After unclamping the shield plug and purging the unit with argon, the pump and its shield plug are raised through the gate valves by an internal hoist. Both of the gate valves are then closed, and the container is raised and

transferred to the servicing room. Replacement is accomplished in a like manner.

The pump unit which is replaced consists of the entire pump, shaft, bearings, and shield plug, as well as the outlet check valve. Connection at the lower end is made by a sealed fitting in which the seal members are removed from a cylindrical hole. All components which may require servicing are a part of this removable unit.

Removal of an intermediate heat exchanger presents a particular problem, since secondary sodium must be at least partly drained and the pipes must be cut before installing a transfer container. Drainage of secondary sodium below the level of the primary sodium may be undesirable if a leak is present, since this would permit the entry of contaminated primary sodium into the secondary side of the heat exchanger. Except for these differences, the intermediate heat exchangers would be removed in the same way as the primary sodium pumps.

3.2 SECONDARY HEAT TRANSFER SYSTEM

3.2.1 Introduction

3.2.1.1 System Function

The main function of the secondary heat transfer system is to transfer energy from the primary heat transfer system to the main steam system.

3.2.1.2 Summary Description

The secondary heat transfer system (see Figure 3.5) consists of three independent heat transfer circuits which are filled with molten sodium. In each circuit the sodium is forced through the shell side of an intermediate heat exchanger by a centrifugal pump. The sodium is heated as it passes vertically

upward through the intermediate heat exchanger. The heated sodium is conducted to the steam generating equipment.

At the steam generators, the heated sodium is split into two streams. One stream flows through the shell side of a steam superheater, and the other stream flows through the shell side of a steam reheater. The sodium is cooled somewhat as it passes through those units.

The two streams are then mixed in a header at the base of the superheater and the reheater. The mixed stream is conducted vertically upward to the top of the evaporators. The sodium is split into two streams which flow in parallel through the shell sides of two identical water evaporators. The sodium is cooled further as it passes through the evaporators.

The cooled sodium is conducted to the suction side of the sodium pump.

The steam generators are located at a higher elevation than are the intermediate heat exchangers and the primary heat transfer system. This assures that the static pressure on the secondary side of the intermediate heat exchangers exceeds that on the primary side. Consequently, the highly radioactive primary sodium will not leak into the secondary sodium. Further, the elevation difference aids heat removal by natural circulation of the secondary sodium if electrical power to the secondary sodium pumps is lost.

Thermal expansion of the sodium in the secondary heat transfer system is accommodated in an expansion tank. The tank is located at the highest elevation in the system. It is connected to the system at the point of least pressure fluctuation which is just upstream of the pump inlet. Two smaller tanks are provided at two high spots in the system to facilitate draining.

3.2.1.3 Design Requirements

The secondary heat transfer system is required to utilize liquid sodium to transport heat from the IHX. Over the normal range of plant operation, the secondary heat transfer system must be capable of smoothly varying the sodium flow rate in response to the plant control system (Section 3.13). The following are specific requirements of the secondary heat transfer system:

1. The flow rate of sodium must be smoothly variable over the range of normal operation.
2. Either forced or natural circulation of sodium must be assured under all emergency conditions.
3. No throttling valves or tight shut-off valves are to be used to control the sodium flow.
4. The pressure of sodium on the secondary side of the IHX must be greater than the pressure of the primary heat transfer system.
5. The reliability of the components must be such that the plant unavailability is less than 10%:
 - a. Unavailability due to component failure must be less than 5%, and
 - b. Unavailability due to instrumentation failure must be less than 5%.
6. The radioactivity of the secondary sodium must be less than that which would produce a radiation dose of 7.5 mR/h at three feet from the surface of the large sodium pipes.

FIGURE 3.5 SECONDARY HEAT TRANSFER SYSTEM
(SIMILAR TO GE DRAWING 277E226)

3.2.2 Detailed Description

3.2.2.1 Steam Generators (R. G. Stuart)

The steam generators are required to produce steam at the lowest possible cost consistent with safety. A high availability, which results in a high load factor throughout the thirty-five years design life of the plant, is also required. A high availability can only be achieved by minimizing the probability of accidental mixing of the water and sodium owing to component failure.

These two basic requirements result in the following set of design criteria for the steam generator units.

1. The steam generators shall be of the shell and tube type with the water or steam within the tubes and the sodium within the shell.
2. The tube sheets shall be separated from contact with the sodium by a gas space.
3. The shell shall be protected against thermal shock.
4. The tubes shall be free to expand relative to the shell and to each other.
5. The design shall incorporate features that produce a high integrity barrier at the sodium and water interface and that minimize the risk of a leak.
6. The design shall incorporate features that minimize the damage resulting from a leak between either sodium and steam or sodium and water.
7. Adequately uniform flow distribution within the steam generator of both water (or steam) and sodium shall be ensured over its full operating range.
8. Gas entrainment by the sodium shall be limited to a level at which its effect on operation or performance of the steam generator is reasonably small.
9. The design shall not allow any damaging vibrations of the tubes or tube spacers.

10. The units shall be fully drainable on the sodium side.
11. The design shall provide adequate access for inspection and maintenance.

Figures 3.6 through 3.9 illustrate the design features of a typical steam superheater unit. The dimensions shown on the figures are superseded by those shown in Table 3.3. The design features are typical also of both the evaporator and the reheater units. The dimensions of the evaporator and the reheater units are shown in Table 3.3.

As shown in Figure 3.6, each steam generator unit appears externally as a vertical cylindrical shell. The heat transfer surfaces take the form of "U"-tubes. The choice of U-tubes meets the objectives of this study. Another choice may eventually prove to be more attractive. A discussion of alternatives is presented in the Task IV report.

The vertical legs of the tubes are closely spaced in two annular passages. The pitching of the tubes is such that the tube ends form an involute pattern at the tube sheets. The arrangement of the tubes is shown on Figure 3.7.

The U-tubes are arranged to connect two concentric annular tube sheets (see Figure 3.8). Water or steam enters at the inner tube sheet and leaves at the outer tube sheet. These tubes are free to expand relative to the shell and to each other. The long connecting section at the bottom of the U allows for relative expansion between the two legs of the U-tube.

Hot sodium enters the steam generator (either the superheater, the reheater, or the evaporator) by a single thermal sleeve connection to a distribution ring which is placed just above the main shell flange. From the distribution ring, sodium distribution nozzles pass the sodium into the flow passages of the

TABLE 3.3

STEAM GENERATOR DATA

Type	Once-through			
Shell-Side Fluid	Sodium			
Tube-Side Fluid	Water or Steam			
Physical Description	Evaporator	Superheater	Reheater	
Number	6	3	3	
Number of Tubes	759	439	345	
Tube o.d., inches	0.944	1.225	1.863	
Tube Thickness, inches	0.1597	0.175	0.0567	
Corrosion Allowance, inches	0.030	0.030	0.030	
Total Tube Length, feet	110.3	72.1	71.1	
Inactive Tube Length, feet	15.0	15.0	15.0	
Tube and Shell Head Material	2-1/4 Cr-1 Mo	316 SS	316 SS	
Shell Material	2-1/4 Cr-1 Mo	304 SS	304 SS	
Operating Conditions (Each Unit)				
Heat Transfer Duty, MW(t)	280	118	127	
Active Heat Transfer Area (o.d.), ft ²	17,880	8,040	9,440	
Water or Steam Flow, lb/h	1.2×10^6	2.4×10^6	2.17×10^6	
Sodium Flow, lb/h	10.7×10^6	10.3×10^6	11.07×10^6	
Sodium Inlet Temperature, ° F	942.3	1071.4	1071.4	
Sodium Outlet Temperature, ° F	644.9	942.3	942.3	
Sodium Pressure Drop, psi	34.2	22.4	22.4	
Water or Steam Inlet Temperature, ° F	480.7	762.5	600	
Steam Outlet Temperature, ° F	762.5	950	950	
Water or Steam Inlet Pressure, psia	2581	2576	579	
Steam Outlet Pressure, psia	2576	2533	570	
Heat Transfer Rate, Btu/h-ft ² -° F (o.d.)	*	335	217	
Mean Temperature Difference, ° F	*	148.6	212.7	
*Evaporator Heat Transfer Regions	Sub-Cooled	Nucleate	Film	Superheat
Heat Transfer Duty, MW(t)	97	60	60	63
Heat Transfer Area (o.d.), ft ²	7,650	3,690	2,870	3,560
Sodium Inlet Temperature, ° F	747.8	811.8	876.0	942.3
Sodium Outlet Temperature, ° F	644.9	747.8	811.8	876.0
Water or Steam Inlet Temperature, ° F	480.7	672.5	672.4	672.3
Steam Outlet Temperature, ° F	672.5	672.4	672.3	762.5
Heat Transfer Rate, Btu/h-ft ² -° F (o.d.)	380	535	422	316
Mean Temperature Difference, ° F	113.9	103.8	169.1	191.2

FIGURE 3.6 STEAM GENERATOR-ELEVATION
(SIMILAR TO GE DRAWING 277E173)

FIGURE 3.7 STEAM GENERATOR—LOWER SECTION
 (SIMILAR TO GE DRAWING 277E180)

FOR STEAM GENERATOR ASSEMBLY, SEE
DRAWING NO. 277E173.

FIGURE 3.8 STEAM GENERATOR UPPER END-PLAN
VIEW (SIMILAR TO GE DRAWING
277E140 SH. 1)

FIGURE 3.9 STEAM GENERATOR UPPER END-ELEVATION (SIMILAR TO GE DRAWING 277E140 SH. 2)

outer annulus below the normal operating level (see Figure 3.6). One sodium distribution nozzle is located in the space between each pair of tubes in the outermost row of each steam generator unit. Sodium then passes vertically downward, parallel to the tubes, through the outer flow annulus. At the bottom of the shell, the sodium turns and flows vertically upward along the tubes in the inner flow annulus. Before reaching the top of the U-tube, the sodium flow turns and flows vertically downward through a central exit pipe. The sodium finally leaves the steam generator unit at the bottom of the shell. The top of the exit pipe is below the operating sodium level in the inner flow annulus.

The outer and inner sodium flow paths are separated by an annular baffle. Stagnant sodium will partially fill the space between the two parts of the baffle. The level of sodium in the interpass baffle will decrease with increasing flow rate. The space in the baffle, above the sodium, is filled with argon cover gas.

Feed water or steam enters four headers, symmetrically spaced around the inner tube sheet (see Figures 3.8 and 3.9), from the main feed pipes or from the evaporator to superheater connecting pipes or from the high pressure turbine exhaust pipes. The steam or water enters the tubes after passing from the headers to the inner tube plate. The steam then flows vertically downward into legs of the U-tubes which are located in the inner annulus. At the bottom of the down flow leg, the steam or water is turned and flows up the leg of the tube which is located in the outer annulus. From the tubes, the steam exits to the outer tube plate and then to four symmetrically spaced headers. Orifices are provided at the inlet tube sheet to enhance flow distribution and stability. This flow path arrangement results in the two fluids being in counterflow, thus obtaining the maximum usage of the heat transfer surface.

Lateral tube spacing and tube support is provided by spacers in the sodium stream. The spacers are not intended to make the sodium flow deviate from its vertical downward or upward path.

Because the main shell is the principal containment, it is protected from possible thermal shock by an insulating gas space. This gas space is contained between the shell wall and a thinner containing wall which is joined to the sodium outlet pipe at the bottom of the vessel. In the unlikely event of sodium spilling into the gas space, a drain is provided at the bottom of the vessel. For removal of the tube bundle for necessary repairs and for initial assembly of the steam generator, the main shell is divided into two parts which are held together by a flange. The flange is located just below the sodium operating level. A connection is provided at the bottom of the vessel for both filling and draining the sodium.

Shell and tube heat exchangers are generally susceptible to leaks at the tube to tube sheet welds. This susceptibility is aggravated by subjecting the welds to thermal shock. Adequate protection against thermal shock is provided by insulating gas spaces which separate both the inner and outer tube sheets from the sodium.

In the event of a tube failure which leads to a sodium-water reaction, hydrogen, sodium oxide, and sodium hydroxide will be formed. The hydrogen will increase the pressure in the cover gas space and present a potential hazard to the main steam generator containment vessel. In order to detect hydrogen release, hydrogen detectors and pressure sensors are provided in each gas space. Also, bursting disks are provided for each gas space. Their bursting pressure is set to be approximately 150 psig, which is 200 psi below the design pressure of the main containment shell.

A more detailed description of this pressure relief system is presented in Section 3.6.

The start up of the steam generator system is an adaptation of fossil fired plant start up procedures. The start up procedure makes use of a starting vessel or steam separator and an auxiliary steam supply for quick starting. Use of the starting vessel or steam and water separator converts the once-through system into a recirculating system during start up. At the pressure and temperature at which the turbines are loaded, the system reverts to the once through mode. The plant start up procedure is described in Section 3.13.

During shutdown periods it will be possible to inspect the steam generator units on the tube side through openings provided at the top of the steam water headers. The headers are sealed by cover plates during operation (Figure 3.8). The headers and tube sheets may be visually inspected, and using an introscope, the tubes may be inspected along their straight lengths. The sodium will be drained from the units before inspection. It is assumed that no attempt will be made to inspect the shell side of the steam generator units unless there is a component failure necessitating removal of the tube bundles or the replacement of a bursting disk.

During inspection it is important that the tubes be kept warm, say 150° F, and trays containing dessicant must be fitted into each header to prevent ingress of moisture. A thorough purge with dry nitrogen should be made during the inspection.

Yearly inspection would indicate whether the build up of solids require that the inner tube walls should be flushed and cleaned. Acid cleaning can be used with the following provisions:

- a. The acid solution should have a carefully chosen inhibitor so that a minimum amount of metal is removed.
- b. The heating surface should be flushed thoroughly with clean water and a carefully chosen neutralizer. Caustic compounds should not be used.
- c. A final rinse with demineralized water should be used.
- d. A purge with dry nitrogen should be used to remove traces of flushing water.

Once the tubes have been acid cleaned, the inner tube surfaces could be further inspected for possible pitting corrosion and other forms of corrosion. If the inspection showed that corrosion is severe, it may be possible to perform an eddy current test for the detection of possible cracks.

Table 3.3 is a summary of the characteristics of the steam generator units.

3.2.2.2 Secondary Sodium Pumps and Drives (C.E. Boardman)

Three parallel secondary sodium coolant pumps are located at ground level in the steam generator building. The layout of the steam generator building is described in Section IV. Each pump circulates secondary sodium through its own independent loop which consists of the shell side of an intermediate heat exchanger, the shell side of the steam generator units, and the connecting piping. Each of the secondary coolant pumps is a centrifugal-type, free-surface, single suction pump with a removable inner barrel or casing. Those units are designed to utilize a sodium-lubricated self-aligning hydrostatic bearing, gas lubricated face-type seals, and a vertical would-rotor drive motor with a liquid rheostat speed control. The choice of a would-rotor motor meets the objectives of this study. Another choice may eventually prove to be more attractive. A discussion of alternatives is presented in the Task III report.

Each pump and its motor is supported at the lower flange of the motor. Each pump extends vertically downward inside a concentric support cylinder (outer barrel). The outer barrel contains the removable portion of the pump unit and the sodium pool from which the pump takes suction. The pump discharge pipe is sealed to the discharge nozzle, which is an extension of the outer barrel, by a sliding fit labyrinth seal. This arrangement allows for vertical removal of the pump for maintenance. The inner removable barrel or housing is required to support the radial hydrostatic shaft bearing, the volute case, and the discharge piping. The inner barrel is a cylindrical shell from the support flange down to the volute case. The only perforations in the barrel are those above the radial bearing which allow bearing overflow to join the sodium pool in the outer barrel.

The pump shaft length from the top coupling to the pump impeller is 14 feet and the overall withdrawal length down to the end of the discharge nozzle is 20 feet. During normal three-loop operation, each pump will operate at a balanced 49,076 gpm flow with a total developed head of 243 feet. Performance and design information are presented in Table 3.4.

The pump shaft is supported by an oil lubricated thrust bearing located within the sealing unit. Shaft alignment is maintained by a self-aligning radial hydrostatic bearing located in the sodium pool. High pressure sodium from the pump discharge is supplied to the bearing to maintain impeller concentricity. The bearing overflow is allowed to leak back through perforations in the inner barrel to the sodium pool in the outer barrel.

An oil seal unit provides lubrication and cooling for the pump thrust bearing and mechanical seals. The seals are gas tight and isolate the sodium pool argon cover gas

from the steam generator building air zone. Each secondary pump unit is driven by a vertical 3400-hp, 1200-rpm (synchronous speed), three-phase, wound-rotor motor. Each motor has a liquid rheostat to provide speed control from 20% to 100% of rated pump speed.

3.2.2.3 Piping and Expansion Tanks

Tables 3.5 and 3.6 contain a summary description of the secondary sodium piping and of the secondary sodium expansion tanks, respectively.

3.2.3 Principles of Operation

The piping and components of the secondary heat transfer system may be heated and maintained at 350° F by externally mounted resistance heaters. The preheated secondary system can be filled by the sodium service system (Section 3.5). Sodium at 350° F is supplied through the small diameter drains at the low points of the system.

When the reactor system is not in operation, the sodium is circulated isothermally through the secondary heat transfer system. The secondary sodium pumps are driven at 20% of full speed by the low power pony motors. Any decay heat from the primary heat transfer system is transported to the plant start-up system (Section 3.13).

When the reactor system is being brought up to full power, the flow rate of sodium flow through the secondary heat transfer system is smoothly increased by the variable speed, wound-rotor motor, main drives of the sodium pumps. A description of the plant start-up procedure is presented in Section 3.13.

During normal operation, the speed of the secondary sodium pumps is varied in response to the plant control system (Section 3.13).

TABLE 3.4

SECONDARY SODIUM PUMP CHARACTERISTICS

Type	Centrifugal, shaft sealed, sump type
Fluid	Non-radioactive molten sodium
Material	Stainless Steel
Number	3
Flow Rate, lbm/h	21.416×10^6
Flow Rate, gpm	49,076
Pump Speed, rpm	1115
Total Developed Head, ft of Na	243
Hydraulic Horsepower, hp	3111
Net Positive Suction Head, ft of Na	130
Operating Temperature, ° F	645
Design Temperature, ° F	850
Cover Gas Pressure, psia	45
Depth of Impeller, ft	7
Shaft Length (from coupling to pump impeller), ft	14
Withdrawal Length	20

Main Pump Drive Characteristics

Drive Type	Vertical, Wound-Rotor, Induction Motor
Speed Control	Liquid Rheostat in Rotor Circuit
Number	3
Synchronous Speed, rpm	1200
Power Rating, hp	3400
Electrical Supply	3 phase, 60 cps, 4000 volts

Auxiliary Pump Drive Characteristics

Drive Type	Squirrel Cage Pony Motor
Number	3
Power Rating, hp	30

TABLE 3.5

SECONDARY SODIUM PIPING DESCRIPTION

Material	304 Stainless Steel
Design Temperature	1100° F
Operating Pressure	150 psig
Sodium Velocity	25 fps
Inside Diameter	30.9 inches
Outside Diameter	32 inches
Heating Method	External Calrods

TABLE 3.6

SECONDARY SODIUM EXPANSION TANK CHARACTERISTICS

Total Number	3
Volume of Sodium in One Secondary Loop	
Superheater	193 ft ³
Reheater	302
Two Evaporators	1140
Piping	1530
Intermediate Heat Exchanger	700
Total	3865 ft ³
Minimum Loop Filling Temperature	300° F
Average Operating Temperature	880° F
Total Volume Change	349 ft ³
Design Volume	698 ft ³ , 5220 gal
Design Temperature	1000° F
Material of Construction	304 Stainless Steel

During an orderly shutdown, the speed of the secondary sodium pumps is decreased smoothly to 20% of full speed. At that speed the pony motors are energized and the main drives are de-energized.

When a reactor scram occurs, the pony motors are immediately energized. The main drives are de-energized at the discretion of the plant operators. The secondary sodium pumps then coast down to 20% of full speed and are maintained at that speed by the pony motors.

3.2.4 Safety Precautions

The following are measures which must be taken to assure the safe operation of the secondary heat transfer system:

1. The sodium pumps must be regularly monitored for vibrations and evidence of cavitation.
2. The bulk sodium temperature must be monitored during load changes to detect abnormal values.
3. Thermocouples on the pump casings must be continuously monitored to detect uneven cooling.
4. The cover gas pressure and the sodium level in the pump must be continuously monitored to prevent flooding of the pump.
5. The radiation level in the secondary sodium must be continuously monitored.
6. The hydrogen concentration in the cover gas of the steam generators must be continuously monitored as an indication of steam tube leaks.

3.2.5 Maintenance

During regular plant shutdown periods the components of the secondary heat trans-

fer system are carefully examined. At this time, minor repairs and adjustments can be made.

To replace a major component, the sodium must be drained from the affected loop. The component is disconnected from the system and lowered to a transfer vehicle on the ground floor of the steam generator building. The component is braced and supported in a vertical position. The transfer vehicle then moves slowly along a transfer aisle provided in the steam generator building. The component is transferred to the turbine auxiliary bay where it can be laid horizontally if necessary for repair.

3.3 STEAM, FEEDWATER, AND MAKEUP WATER SYSTEM (Bechtel Corp.)

3.3.1 Introduction

3.3.1.1 System Functions

The primary function of the steam, feedwater, and makeup water system are to transfer energy from the secondary sodium system to the main electrical generating system and to reject to the atmosphere the energy that is not converted to electricity.

3.3.1.2 Summary Description

The plant (Figure 3.10) utilizes a regenerative, reheat steam cycle. The choice of a reheat cycle meets the objectives of this study. A non-reheat cycle may eventually prove to be more attractive. A comparison of the reheat and non-reheat cycles is presented in the Task III report.

Condensate from the condenser hotwell is heated by steam extracted from the turbines, then pressurized, heated further by extraction steam, and conducted to the boiler inlet. Superheated steam from the boiler is conducted to a separate superheater unit and is

LEGEND
 H, h = ENTHALPY - BTU/LB.
 # = FLOW - LB./HR.
 P = PRESSURE, PSIA
 T = TEMPERATURE, F DEGREES

FIGURE 3.10 PLANT HEAT BALANCE DIAGRAM
 (SIMILAR TO GE DRAWING 106D3629)

heated further. High pressure superheated steam is conducted to the inlet of the high pressure turbines. Some of the steam exhausted from the high pressure turbine is used to preheat the pressurized boiler feedwater. Most of this steam is conducted to a separate reheater unit and is heated to the temperature of steam leaving the superheater. The lower pressure reheated steam is conducted to the intermediate pressure and low pressure turbines. Some steam is extracted from these turbines to preheat the boiler feedwater. Most of the steam from the lower pressure turbines flows to a condenser and is condensed at a pressure less than atmospheric pressure.

Additional high-purity water must be provided to the system to make up for losses.

The condenser is cooled by river water which is pumped through the condenser tubes and returned to the river.

The design requirements for the components of the main steam systems are indicated on the heat balance diagram (Figure 3.10).

3.3.1.3 Design Requirements

The steam, feedwater, and make-up water system is required to remove heat from the evaporators, superheaters, and reheaters during normal operating conditions. The following are specific requirements of the system:

1. The flow rate of steam and water must be smoothly variable over the range of normal operation.
2. The pressure of steam delivered to the turbines must be held nearly constant over the range of normal operation.

3. The reliability of the components must be such that the plant unavailability is less than 10%:
 - a. Unavailability due to component failure must be less than 5%, and
 - b. Unavailability due to instrumentation failure must be less than 5%.
4. The rotational speed of the turbines must be held constant over the range of normal operation.
5. High purity water must be available to continuously replace all leakage from the system.
6. The steam generator and piping must be designed in accordance with the ASME Boiler and Pressure Vessel Code and the ASA Code for Pressure Piping.
7. The system must be capable of changing the turbine shaft power at the rate of 10% of full power per minute in the range from 50% to 100% of full power, and at 5% per minute in the range from 20% to 50%.
8. The steam and feedwater temperatures and pressures are fixed by optimizing all of the major plant parameters.

3.3.2 Detailed Description

3.3.2.1 Condensate and Feedwater Sub-Systems

The turbine exhaust steam is condensed in a multi-pressure, three-compartment condenser. The three sections are inside a single shell, arranged parallel to the turbine

generator shaft. The condenser tubes run through all sections in a continuous length of 80 feet. The hotwell extends the entire length of the condenser and can hold condensate equivalent to about four minutes of full-load operation.

Selection of a multi-pressure condenser is based upon the superior thermodynamic performance and economic advantage of this type of condenser on turbine generators in the 500 to 1000 MWe range. The vacuum in each condenser, or sealed condenser section, is determined by the temperature of cooling water leaving each section of the condenser. The cold inlet water thus produces a lower pressure in the first compartment than it does when it becomes warmer in the succeeding sections. The multi-pressure condenser has a lower average back-pressure than a conventional single-pressure condenser with the same water flow and surface. However, use of multi-pressure condensers results in substantial savings.

Two half-capacity condensate pumps take suction from the highest pressure section of the main condenser hot-well and discharge it to a common header. (See Figure 3.11). The flow is then split into three equal branches which flow through air ejector condensers and then recombine to a single path through the gland seal condenser. Condensate leaving the gland seal condenser passes through a full-flow demineralizer system. This system includes six ion exchanger vessels, five of which accommodate full flow, with the sixth on standby. The polishing demineralizers are of the Powdex type, which discard depleted resin upon back wash. A full-flow bypass is provided around the polishing system; this bypass may be used to control the amount of condensate flow sent through the polishing system.

The condensate system components are described in Table 3.7.

After leaving the demineralizer, the condensate flow is split again into three identical branches, each passing through a closed feedwater-heating train. The first two heaters in each train (heaters 7A, B, and C and 6A, B, and C) are accommodated in the condenser neck. This arrangement shortens the low-pressure extraction piping, and results in a compact and logical equipment layout. Water boxes protrude from the condenser, facilitating maintenance work on the heaters without disrupting the condenser proper. Drain lines of the heaters are also brought outside of the condenser neck, in order to provide access to the drain controllers. Stainless tubing is used for these heaters since they cannot be nitrogen-blanketed while the unit is out of service.

The feedwater system components are described in Table 3.8. After the two lowest-pressure heaters, the condensate is passed to the third extraction heaters (heaters 5A, B and C) located outside the condenser. These use carbon-steel tubing since the shells can be nitrogen-blanketed for corrosion protection when the units are out of service. Extraction steam to each heater is taken from each of the three low-pressure turbine sections. A common bypass is provided for any of the No. 6 and No. 7 lowest-pressure feedwater heaters, in case one of the heaters in the condenser neck is inoperable. For the three extraction low-pressure heaters (5A, B and C), a one-third capacity bypass is provided.

All three trains of high-and low-pressure heaters have integral drain coolers. The drains from the high-pressure feedwater heaters cascade to the deaerator. The low-pressure heaters cascade to the main condenser. There is a bypass provided so that

FIGURE 3.11 CONDENSATE AND FEEDWATER SYSTEM
(SIMILAR TO BECHTEL DRAWING M-11)

TABLE 3.7

DESCRIPTION OF CONDENSATE SYSTEM COMPONENTS

Component	Quantity	Description
Main Condenser	1	Single shell, multi-pressure Surface area 464,000 sq ft Total water flow 326,000 gpm 1 in.-o.d. BWG 18 admiralty tubes 81 ft long Back pressure 1.43 in. Hg (Average of 3 condensers) Cleanliness factor 0.85 Water inlet temperature 60° F Total heat rejection 4,379,300,000 Btu/h
Air Ejector	3	300 lb/h saturated air vapor mixture capacity Twin element, two-stage steam jet type with inter- and after condensers
Air Ejector De-superheater	1	Complete unit to service steam to three air ejectors simultaneously
Condensate Pumps	2	1/2 capacity, 7,000 gpm vertical centrifugal, barrel-type motor-driven, 630 ft TDH
	2	1200 hp motors 4160 volt, 3 phase, 60 cycle, 1185 rpm
Condensate Storage Tank	1	100,000 gal 27 ft dia. x 27 ft high 1/4-in. welded steel plate
Condensate Demineralization System	6	Units, Powdex type Total flow for 5 units, 14,000 gpm Design Pressure 225 psi Unit Flow 4 gpm/sq ft Manual Control
	6	Vessels (1 Unit for Standby)
	1	Precoat tank, pump and agitator
Condensate Transfer Pump	1	700 gpm, 50 ft TDH Centrifugal pump
	1	15 hp motor 480 Volt Three-phase
Gland Seal Condenser	1	Unit with internal bypass cooling water flow 14,000 gpm

TABLE 3.8

DESCRIPTION OF FEEDWATER SYSTEM COMPONENTS

Component	Quantity	Description
Boiler Feed Pumps	2	Turbine-driven multi-stage Barrel-type centrifugal 7800 gpm at 330° F with 134 psia suction and 3000 psia discharge Pump speed 4500 rpm Available NPSH 100 ft static
B. F. P. Drivers	2	Turbines, 167,899 lb/h throttle steam each Steam inlet pressure 137.1 psia 2.0 in. Hg abs. exhaust pressure
B. F. P. and B. F. P. Turbine Lube Oil Conditioning System	2	Complete units, including pumps, coolers, reservoirs, vapor extractors, conditioning units
Deaerator (Feedwater Heater No. 4)	1	Feedwater in - 5,623,400 lb/h at 199.1 Btu/lb Feedwater out - 7,265,400 lb/h at 273.2 Btu/lb Working pressure 70.6 psia Design pressure 100 psig ASME Code Storage, 34,000 gal (11 ft dia. × 50 ft long)

LOW PRESSURE FEEDWATER HEATERS

Heater Number	5	6	7
Number of Heaters	3	3	3
Steam Flow	lb/h 66,470	61,560	115,630
Feedwater Flow	lb/h 1,874,480	1,874,480	1,874,480
Drain Flow	lb/h 66,470	128,040	243,660
Shell Pressure	psia 23.1	11.5	5.5
Feedwater Inlet Temperature	° F 195.0	160.9	92.6
Feedwater Outlet Temperature	° F 230.8	195.0	160.9
Steam Enthalpy	Btu/lb 1191.1	1143.3	1100.6
Drain Enthalpy	Btu/lb 173.1	138.9	70.6
Drain Temp.	° F 205.0	170.9	102.6
Tube Material	Carbon Steel	Stainless Steel	Stainless Steel
Tube Size	5/8 in. - 18 BWG	5/8 in. - 20 BWG	5/8 in. - 20 BWG
Feedwater Velocity at 60° F	fps 6.5	7.0	7.0
Approximate Surface ea.	ft ² 5800	8000	9400
Shell Dimensions	(46-in. dia. × 27 ft)	(48-in. dia. × 36 ft)	(48-in. dia. × 38 ft)
No. Tube Passes	2	2	2

TABLE 3.8 (Cont'd)

HIGH PRESSURE FEEDWATER HEATERS

Heater Number	1	2	3
Number of Heaters	3	3	3
Steam Flow, lb/h	223,990	129,060	66,010
Feedwater Flow, lb/h	2,421,790	2,421,790	2,421,790
Drain Flow, lb/h	223,985	353,043	419,057
Shell Pressure, psia	570.0	266.8	137.1
Feedwater Inlet Temperature ° F	315.0	346.4	486.7
Feedwater Outlet Temperature ° F	480.7	406.7	346.4
Steam Enthalpy, Btu/lb	1291.6	1419.5	1343.8
Drain Enthalpy, Btu/lb	393.1	328.6	295.5
Drain Temperature ° F	416.7	356.4	325.0
Tube Material	Carbon Steel	Carbon Steel	Carbon Steel
Tube Size	5/8 in. - 18 BWG	5/8 in. - 18 BWG	5/8 in. - 18 BWG
Feedwater Velocity at 60° F fps	7.0	7.0	7.0
Approximate Surface ea., ft ²	8000	7800	7600
Shell Dimensions	(50 in. dia. x 32 ft)	(50 in. dia. x 32 ft)	(50 in. dia. x 30 ft)
No. Tube Passes	2	2	2

the high-pressure feedwater heater drains can also enter the condenser when, due to low load, the pressure difference between heater No. 3 and the deaerator is inadequate to induce flow to the deaerator.

The heater drain flows are regulated by drain controllers. The drain control valves are activated from the condensate level in the heater shell and operate on overflow. The bypassing of the deaerator is automatic. When heater No. 3 does not drain into the deaerator and the condensate level rises in spite of the open drain control valve, the bypass valve to the condenser opens automatically.

Vent lines from each heater are piped to the next lower pressure heater and from heater No. 3 to the deaerator.

Low-pressure heater vents are cascaded all the way to the condenser.

Following the No. 5 heaters, the three branches of condensate are joined in one header which terminates at the deaerating heater (heater No. 4). The deaerating heater is located on top of the turbine building, assuring the net positive suction head (NPSH) requirement of the boiler feed pumps. The deaerator is a direct-contact heater receiving extraction steam from the low-pressure turbines. The extraction steam from the three low-pressure sections is joined and piped in a single header to the deaerator. The deaerator drains by gravity into the feedwater storage tank, which can hold two minutes of full-load feedwater requirement between its extreme water levels. From the feedwater storage tank, the condensate flows to the boiler feed pumps.

Two half-capacity, turbine driven, barrel-type, multistage boiler feed pumps provide the required feedwater pressure. These pumps discharge into three parallel lines, each passing through a three-stage high-pressure heater train. Three trains were selected in order to handle the large feedwater flow in heaters with conventional diameters, and to facilitate equipment layout.

After passing through the heater trains, the feedwater is fed to three pairs of evaporators, where steam is generated by transferring secondary sodium heat to the feedwater.

Condensate is used as the cooling medium in the gland steam condenser to condense the steam drawn from the turbine shaft packing glands by the gland exhaust system.

The gland steam condenser is of single pass, straight tube design. The condensate enters the inlet channel, makes one pass through the tubes and leaves through the outlet channel. An inner bypass is provided, splitting the condensate flow to the gland steam condenser tubes.

The gland steam condenser is maintained at a slight vacuum through two locally controlled blowers, one of which is continuously operating while the other is on stand-by. The blowers discharge air and non-condensables through non-return valves and a common header to atmosphere.

The gland steam control valve maintains constant pressure in the gland seal system by supplying auxiliary steam or by leaking excess steam to the gland steam condenser. From the gland seal regulator, steam flows to an equalizing header provided with branches which feed the respective turbine cylinder glands as required. Excess steam from the high-pressure turbine glands

flows to the equalizing header when the turbine is at higher loads. A relief valve and continuous train connections are provided on the equalizing header for the protection of the system.

3.3.2.2 Steam Generation And Extraction Subsystems

Steam is generated in six once-through evaporators and superheated in three superheater units. The main steam, leaving the superheaters, is carried to the high-pressure turbine stop valves by two heavy-wall seamless pipe lines. Allowable stresses conforming to the ASME Boiler and Pressure Vessel Code and materials commonly used in critical piping for similar purposes in conventional power plants are used. Steam velocities are limited to 200 fps. The two lines leaving the high-pressure turbine split into three "cold reheat" lines which carry the steam to three reheaters (Figure 3.12). The pipe diameters for these lines are large, but the pipe walls are thin because of the lower steam pressure. Steam supply lines for the highest pressure feedwater heaters (heaters 1A, B and C) branch from the cold reheat lines. Four "hot reheat" lines carry the steam from the reheaters to the intermediate pressure turbine. The selection of four smaller lines in lieu of three larger ones was based on layout requirements. The four lines are headered into two lines at the reheater stop valves. From the intermediate pressure turbine, two crossover lines carry the steam to the three low-pressure turbines.

The main steam system piping is described in Table 3.9. Extraction steam piping is sized according to current design practices. Superheated steam velocities are limited to 200 fps; saturated steam velocities to the lowest pressure heaters (heaters 6A, B and C and 7A, B and C) are limited to 100 fps.

TABLE 3.9

MAIN STEAM SYSTEM PIPING DESCRIPTION**MAIN STEAM**

Designed per ASME Boiler and Pressure Vessel
Code, Section I, 1967

Material - SA - 355 P-22 (2-1/2 Cr - 1 Mo.)

Design Pressure = 2520 psia

Design Temperature = 960° F

Steam velocities vary from 170 to 180 fps
depending upon pipe size.

14 in. pipe	I. D. = 11.00 in. (Evap. to Super- heat Lines Design Temperature 775° F)
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16 in. pipe	I. D. = 12.00 in.
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18 in. pipe	I. D. = 13.75 in.
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24 in. pipe	I. D. = 18.50 in.
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COLD REHEAT

Designed per ASA B 31.1 - 1955 Code for Pressure
Piping

Material, A155 Grade KC 70, Class I

Design Pressure = 600 psia

Design Temperature = 600° F

Steam velocity = 150 fps

28 in. pipe	I. D. = 26.75 in.
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30 in. pipe	I. D. = 28.75 in. (Velocity = 190 fps for short distance)
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HOT REHEAT

Designed per ASME Boiler and Pressure Vessel
Code, Section I, 1967

Material - SA - 335 P11 Seamless (1-1/4 Cr -
1-1/2 Mo.)

Design Pressure = 530 psia

Design Temperature = 950° F

Steam velocities vary from 260 to 280 fps depending
upon pipe size

24 in. pipe	I. D. = 22.50 in.
-------------	-------------------

28 in. pipe	I. D. = 26.50 in.
-------------	-------------------

FEEDWATER

Designed per ASA B31.1 - 1955 Code for Pressure
Piping

Material - A106 Grade B Seamless, Carbon Steel

Design Pressure = 3000 psia

Design Temperature = 330° F to 500° F

Discharge line velocities vary from 19 to 23 fps
depending upon pipe size

Suction line velocity = 4 fps

10 in. pipe	SCH 160	I. D. = 8.5 in.
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14 in. pipe	Special	I. D. = 11.0 in.
-------------	---------	------------------

TABLE 3.9 (Cont'd)

FEEDWATER (Continued)	16 in. pipe	Special	I.D. = 12.625 in.
	24 in. pipe	Special	I.D. = 19.0 in.
	30 in. pipe	SCH 10	I.D. = 29.376 in. (Suction Line)
EXTRACTION	Designed per ASA B 31.1 - 1955 Code for Pressure Piping		
	Material - A106 Grade B Seamless, Carbon Steel		
	Design Pressure and Temperature depending on extraction points		
	Steam velocities limited to 200 fps for superheated steam, and 100 fps for saturated steam.		
	1/2-in. to 10-in. pipe - Sch 40		
	10 in. and above - 0.375 in.		

High-pressure feedwater heaters 2A, B, and C and 3A, B, and C receive steam from the intermediate pressure turbine through two exit nozzles which are split into three steam lines, each leading to a high-pressure feedwater heater.

The deaerator (heater No. 4) and the low-pressure feedwater heaters receive extraction steam from the three low-pressure turbines; each heater train is supplied by one of the turbine cylinders. Steam is supplied to the main air ejector condenser through lines which branch from each one of the three cold reheat lines.

Steam takeoff is provided for the boiler feed-pump turbine drives from the crossover lines between the intermediate and low-pressure turbines. The boiler feed-pump turbines exhaust into the highest pressure main condenser section.

3.3.2.3 Turbine Generator Subsystem

a. Turbine Description

The turbine is a tandem-compound reheat unit rotating at 3600 rpm.

The turbine consists of five sections: a high-pressure section, a double-flow re-

heat section, and three double-flow low-pressure sections. Steam passes through the four control valves, which are in a separately mounted valve chest, enters the turbine at the generator end of the high-pressure section and flows toward the front standard. After passing through the high-pressure stages, the steam is returned to the reheater. The reheated steam returns to the turbine through the combined reheat stop valves and intercept valves. Upon entering the center of the reheat section the steam divides, part flowing toward the generator end of the section and part toward the turbine end. The steam leaves the reheat section through the crossover pipes which connect to the inlet of the low-pressure elements. One third of the steam enters each double-flow section. After passing through the low-pressure stages, the steam is exhausted downward into the condensers.

The high-pressure section is made with double-shell construction. Steam at the high-pressure section exhaust pressure is injected between the outer and inner shells, resulting in simpler and more stable shell castings. The inner shell is located in the outer shell in the axial direction by a rabbit fit, and horizontally by gib fits on the upper and lower vertical centerlines at each end

of the inner shell. The inner shell is supported vertically on four support pads. The shims used on the supporting pads, to secure correct vertical alignment of the inner shell, have hardened surfaces to eliminate the wear which may result from the relative movement of the inner shell as it expands and contracts.

Each of the control valves in the separate control valve chest regulates the flow to one of the four nozzle boxes. The nozzle boxes contain the high-pressure, high-temperature steam until it passes through the first-stage nozzle openings. The nozzle boxes are made with relatively thin walls so that they heat uniformly when the hot steam is introduced. The nozzle boxes are located in the inner shell by keys and a rabbet fit which permit the boxes to expand freely, yet be located accurately with respect to the first-stage wheel and buckets under all operating conditions. The nozzle boxes, combined with the double-shell construction, minimize thermal and pressure stresses in the turbine shells in the area of the first stage. In addition, the two sets of nozzle openings in the nozzle boxes balance out the steam reaction forces which reduces the stresses on the nozzle box locating keys.

The nozzle box steam inlets which extend into openings in the outer shell are equipped with slip-ring expansion joints which permit the nozzle boxes to move in any direction with respect to the shells and still maintain a steam-tight fit. Similar expansion joints are used for the reheat inlets in the reheat section.

The reheat section also has double-shell construction in the area of the reheat inlets. The inner shell, at the reheat inlet, is supported and held in alignment in a manner similar to that used for the high-pressure inner shell. Since the reheat section is double-flow and the steam divides and

flows in both directions from the center, this section of the turbine is symmetrical.

The outer shells of both the high-pressure and reheat sections are supported by shell arms which extend onto the standards. The high-pressure shells are supported by the front standard and the second standard, and the reheat shells are supported by the second and the third standards. The shells are supported at their horizontal centerlines, on both ends, and are keyed to the standards at the upper and lower vertical centerlines to prevent transverse movement. This arrangement maintains accurate alignment between the stationary and rotating parts as the shells expand and contract under varying operating conditions.

The turbine is keyed at the exhaust hood to prevent transverse movement and the turbine expands from this point. The first and the second standards are free to slide in the axial direction but are guided to prevent transverse movement.

The three double-flow, low-pressure sections are similar in construction. Each has a separate inner casing, which is supported in the outer hood by four support pads, and which is keyed to prevent axial and transverse motion. The steam inlet is provided with a bellows expansion joint between the inner casing and outer hood which permits the inner casing to move with respect to the outer, and which prevents air leakage into the condenser.

The last hood section has provisions for the turning gear at its generator end.

The turbine has five rotors, each supported on its own bearings. The coupling between the high-pressure and reheat rotors is solidly bolted.

The rotors are located axially by the thrust bearing which is located in the second standard.

Each rotor body is made from a solid alloy steel forging, which is machined to form a solid rotor composed of shaft, wheels, bearing journals and coupling flange. Prior to machining, various tests are made to assure that each forging meets the required physical and metallurgical properties.

The buckets are machined from a chrome-iron alloy that is extremely resistant to corrosion and erosion by steam. They are dovetailed to the wheel rims by a tight machine fit and tied together in segments by punched shroud bands fitted over the tenons on the bucket outer tips. The tenons are then hand-riveted to hold the bands in place.

On last-stage buckets, where the tip speed is high, hardened shields are attached to the upper portion of each bucket as an additional safeguard against erosion due to moisture.

The steam flow is directed against the buckets at the proper angle and velocity by the diaphragm partitions. The nozzle areas and angles of discharge are determined by many variables, such as the volume of steam to be passed, steam pressure drop across the diaphragms, and the velocity of the adjacent buckets.

The nozzle partitions are machined from solid chrome-iron alloy and are incorporated into the diaphragm by either a welding or a "cast-in" process. In high-pressure areas, welded-type nozzle diaphragms are used. The machined, chrome-iron alloy partitions are assembled in punched steel bands and tack-welded into position. This assembly is then welded to the diaphragm web and to the outer

ring. The completed fabrication is then accurately machined.

b. Turbine Control System

The Electrohydraulic Control System (EHC) controls the turbine-generator by positioning the turbine's stop valves, control valves, reheat stop valves, and intercept valves to meet operating conditions. This system represents a substantial advance in turbine control philosophy. The flexibility, fast response, and complex logic capabilities of solid-state electronics are combined with high-pressure, hydraulic actuation to provide a significant degree of control automation and improved operation.

The electrohydraulic control system consists of the following major components:

1. The EHC Cabinet contains the electrical analog and logic circuits as well as the monitor panel and electrical power supplies.
2. The Operating Panel contains the starting and operating controls for the turbine, including indicating instruments, lights and remote test stations for steam valves and the overspeed trip system.
3. The High-pressure Hydraulic System consists of the hydraulic power unit located below the turbine, the valve servomotors, and the hydraulic emergency trip system. High-pressure (1600 to 2000 psi) fire-resistant fluid is used in this system, which is entirely separate from the bearing lubrication system.
4. Control Components at the Turbine Front Standard are; the speed pick-ups, the hydraulic emergency trip devices, and the permanent magnet generator (PMG).

c. Turbine Oil Pumping System

The main oil pump, which supplies oil to the bearings of the turbine-generator shaft, is a centrifugal pump mounted on the turbine shaft. It is supplied with suction oil at about 15 to 25 psig by an oil-driven booster pump located in the oil tank. Oil discharging from the main oil pump is piped to the oil tank where it passes through the oil turbine which drives the booster pump. In passing through the oil turbine, the oil pressure is reduced to about 45 to 55 psig to provide the power to drive the booster pump. The oil then goes through the oil cooler and on to the bearings.

Two motor-driven centrifugal oil pumps are provided to supply bearing oil to the turbine bearings while the turbine is on turning gear, while the turbine is coming up to speed, or in an emergency condition. These pumps are the turning-gear oil pump and the emergency bearing oil pump. They take suction oil directly from the oil tank and discharge it into the bearing header prior to the main oil coolers.

One ac motor-driven centrifugal oil pump is provided to perform the function of the booster pump until the turbine shaft has reached approximately 90 percent of rated speed. This pump is called the motor suction pump.

Because this pumping system contains no motor-driven auxiliary oil pump, high-pressure operating oil is not available to drive the booster pump until the main oil pump has reached about 90 percent of rated speed. Thus, until this speed is reached, the motor suction pump must provide the main oil pump with suction oil at a positive pressure, thus taking over the function of the booster pump.

d. Steam Seal System

The steam space of each turbine section is sealed against steam leakage to the atmosphere where the shaft passes through the shell. In cases where adjacent sections are located in the same shell the shaft is sealed against internal steam leakage from one section to the other. This is done by means of labyrinth-type shaft packings which provide a series of throttlings that limit steam leakage along the rotating shaft to a minimum as it is throttled from the high-pressure space to the low-pressure space.

Pressure packings are those that seal against a positive pressure at full load. At light loads, when a vacuum exists throughout the turbine, they seal against air leakage into the turbine. Pressure packings have a so-called atmospheric leakoff which is held between 2-1/2 and 4-1/2 psig by an air operated feed and a direct acting unloading valve. Steam must be fed into the packing through this leakoff line at light load. At higher loads, when the stage pressures build up, steam flows from the packing into the atmospheric leakoff. Pressure packings may also have one or more intermediate leakoffs which discharge steam to higher-pressure extraction points.

Beyond the atmospheric leakoff annulus there is a vent annulus which is maintained at a slight vacuum (3 and 5 in. H_2O) by a steam packing exhaustor. This takes steam leaking from the leakoff annulus and air pulled in from the atmosphere. The steam packing exhaustor consists of a shell and tube heat exchanger to condense the steam, and a motor driven blower to handle the air and hold a slight vacuum on the system. It is normally supplied with water from the condensate system. Should a heat exchanger tube develop a leak, operation of this system

can continue with the heat exchanger out of service for maintenance. The steam-air mixture is desuperheated and the vacuum is maintained by the blower, which operates with the discharge valve wide open for this condition.

e. Generator Description

The generator is completely enclosed to operate with hydrogen gas as the principle cooling medium. A separate liquid cooling system is provided for the armature windings. The hydrogen cooling system, including the fans and gas coolers, is completely contained within the generator.

The generator casing and the end shields at either end of the casing are of welded construction and support the generator bearings and shaft seals. The all-welded construction provides a hydrogen-tight enclosure.

The stationary armature is attached to and supported by the generator casing. The generator is designed for continuous operation, and is constructed to withstand, usually without harmful effects, conditions incidental to operation such as suddenly applied loads or three-phase short circuits.

The generator auxiliaries provide control and supervision of the hydrogen pressure, purity, armature cooling liquid, shaft-seal oil, and temperatures at critical points in the generator and cooling system.

The stator frame is a cylindrical casing of welded plate construction reinforced internally with axial and circumferential plates. Heavy steel rings are welded to the circumferential plates to stiffen the frame and to serve as supports for the spring mounting which in turn supports the armature core. The frame structure also serves as the support and enclosure for the gas coolers.

The stator frame is supported from the foundation by feet bolted to the sides of the frame. The end shields which contain the generator bearings are bolted to the ends of the stator frame.

The continued increase in generator ratings requires splits in the stator frame structure which contains and supports the magnetic core and windings so that the largest piece does not exceed shipping limitations. The sectional-type stator frame is designed to allow assembly of the core and windings in a portion of the outer frame structure, from which the end sections of the frame can be removed for shipment. The end-frame sections contain the vertical hydrogen coolers, and are assembled to the mid-section at the factory to verify the fits and to allow the generator to be tested, if required. They are removed for shipment, and then re-assembled in the station. This three-section arrangement provides protection and convenience in handling the core section during shipment and installation of very large generators, and holds to a minimum the number of subassemblies involved. It is used for generators on the order of one million kVA and above. The range of ratings to which this applies varies with site location and rail clearances involved.

The spring mounting consists of longitudinal spring bars which provide the stator casing with radial and tangential isolation from the magnetic vibration of the stator core. The armature punchings are assembled on key bars which in turn are attached to the spring bars. The spring bars are bolted to the frame and support the core assembly around its entire length and periphery. The flexibility of the spring bars, which is limited to safe amounts by construction, effectively isolates the core vibration from the stator frame.

The armature core is built up of segmental annealed insulated punchings of

oriented grain silicon sheet steel. The punchings are assembled in an interleaved manner on keys machined integral with the key bars and separated in packages by space blocks to provide ventilation. Each punching contains radial slots for the armature bars, dovetail slots for armature bar wedges, and other dovetail slots for assembly and locking on the key bars. The assembled punchings are clamped into a stiff cylindrical core by pressure applied through end flanges by means of nuts on the stator key bars. Pressure is applied to the teeth of the punchings by means of nonmagnetic steel fingers located under the end flanges. In order to reduce heating from end leakage flux and the associated electrical losses occurring at the ends of the stator core, the end packages of punchings are stepped back to increase the gap between the punchings and the rotor. The punching insulation is a thermosetting varnish containing Santocel, which retains its insulating value after being exposed to abnormal heating.

The armature winding is formed by insulated bars assembled in the stator core slots, joined at the ends to form coils and connected in the proper phase belts by bus rings. The stator bars are composed of hollow, insulated copper conductors (strands) arranged in the form of rectangular bars in which each conductor or strand is so assembled that it occupies, for an equal length along the bar, every radial position in the bar. This arrangement avoids eddy current losses under load conditions which would otherwise be caused by the self-inductive magnetic flux distribution in the coil slot.

Micapal insulation is applied to the assembled bars. Micapal consists of several layers of micamat tape and micaflake tape impregnated with a thermosetting binder which is applied to the bar at several different stages of the taping process. The bars are then placed in a tank and subjected to heating, vacuum and pressure cycles

to remove volatile materials and voids from the insulation and cure the binder. An outer covering of asbestos is then applied on the slot portion to protect the bar from abrasion in the slot. To minimize the effect of corona, semi-conducting paint (containing graphite) is applied to the slot portion of the bar, with a higher resistance grading extending several inches beyond the core.

Non-metallic ripple springs are assembled along one side of the armature bars in the slots. The armature bars are held in the slots by Textolite wedges driven into the dovetail slots near the inner edges of the coil slots.

The end turns of the armature winding are securely laced to non-metallic glass reinforced binding bands with treated glass cord. Conforming plastic blocks, cured after assembly, are used between bars and along supporting surfaces. Groups of bars are held together by fibre glass straps impregnated with a thermal-setting resin, which is also cured after assembly.

The strands of the insulated armature bar are hollow to permit circulation of the cooling water, and to provide intimate contact between the water and the conductor. The strands of each coil are manifolded at the end turns, and connected to the supply and return cooling liquid headers through flexible connectors.

The closed cooling system provided with a water-cooled generator is designed to operate as an independent subloop in the turbine-generator control system. Provision is made for automatic regulation of flow and temperature of clean, low conductivity water to the stator winding.

Included in the main loop are storage tank, two pumps in parallel, two coolers in series, a filter, and the generator. There is a branch bypassing the generator with a

deionizer for continuous purification of a small percentage of the water. In some cases there is a second branch for cooling static rectifiers in the excitation system.

A dished flange of copper is assembled over the nonmagnetic steel clamping flange at each end of the stator core. This serves the purpose of reducing leakage flux from the end of the stator core to the retaining rings on the field, which would cut through the end turns of the stator winding. Losses in the end turns of the stator winding are thereby reduced.

The generator gas coolers are the vertical type, mounted in the generator frame on rails and supports. They are located symmetrically on opposite sides of the generator.

With the coolers secured to the frame, a gasket is compressed between the cooler head and the frame forming a gastight seal. This arrangement permits cleaning the cooler tubes without dismantling the cooler or scavenging the hydrogen from the generator.

A cooler section may be removed in one piece by first disconnecting the cooling water connections and removing the cooler clamps and gaskets. The hydrogen should be completely scavenged from the generator prior to removal of the cooler clamps.

Ventilation of the armature core is accomplished by forcing the cooling gas both inwardly and outwardly through the radial ducts in the core formed by the punchings and space blocks. The circumferential plates in the frame and back of the core, together with the outside wrapper plate, form sections which separate the frame into high- and low-pressure regions through which the cooling gases are forced into or discharged from the stator core. The gas supplied to and discharged

from these sections is conveyed through tubes or ducts which direct the cooling gas from the fans through the machine and back to the fans through the coolers. This arrangement of alternate inward and outward radial gas flow in the stator core results in substantially uniform cooling in the core, and high-voltage bushings, thus reducing temperature stresses in these parts avoiding excessive local heating.

The rotor is machined from forged alloy steel. Prior to machining, extensive tests are made to assure that the forgings meet the required specifications for physical and metallurgical properties. Longitudinal slots, machined radially in the body, contain the field coils. The field coils are held in the slots against centrifugal force by wedges, both magnetic and nonmagnetic types, which are used to obtain proper flux distribution. These wedges are individually fitted and driven into dovetail openings machined in the rotor slots.

The rotor fans, provided for the ventilation of the generator, are the axial flow type. These fans are assembled near each end of the rotor with the entrance and discharge conditions for the gas to and from the fans controlled by vanes and nozzles. The rotor is cooled by the "gap pick-up" method. In this method gas is scooped into the field from the gap through a series of angular holes in the rotor body. Rotation of the field forces the gas into the field, along the winding in an axial direction for a short distance, and then out into the gap again. The end turns of the field winding are cooled by gas forced underneath the retaining rings from the ends, and then discharged from the field body into the gap.

The field winding consists of extruded copper bars formed to provide ventilation channels with maximum cooling surfaces. Multiple turns in one pair of slots and around one pole form a coil. Several coils are as-

sembled around each pole to complete the field winding. Each turn of the winding is insulated from other turns by insulation. The coils are insulated from the slot wall in the body portion by molded slot liners. Molded ring insulation is provided between the coils and the retaining rings, and Textolite or polyester blocking is provided in the end windings to separate and support the coils, and to restrict their movement under stresses due to temperature change and rotational forces.

The end turns are secured against centrifugal forces by retaining rings machined from heat treated alloy steel forgings.

The ALTERREX excitation system for large steam turbine-generators is a reliable method of providing generator d-c excitation by utilizing stationary rectifiers to convert the a-c output of an alternator-exciter driven directly by the main generator rotor.

The d-c output of the rectifier bank goes to the optional main generator field breaker and to the main generator field.

The alternator-exciter field receives d-c power from the voltage regulator thyristor rectifiers which establish proper alternator field voltage in response to either the a-c automatic or the d-c manual regulator components. Power is supplied from the alternator-exciter armature through a step-down transformer.

The ALTERREX excitation system consists of the following major components:

1. The Alternator-exciter is a direct-coupled a-c generator, driven from the main generator rotor. It has a stationary armature and rotating field. The alternator is designed to permit handling as an assembled unit mounted on its own base.

2. The power rectifier system consists of multiple parallel 3-phase diode circuit assemblies, the number depending on the rating of the system. Each circuit is connected to a 5-pole disconnect switch so that the circuit can be completely isolated on both the a-c and d-c sides. Sufficient parallel circuits are provided to allow one complete assembly to be removed from service, and still allow rated field current to be carried on the main generator.

3. The semi-conductor voltage regulator automatically regulates main generator terminal voltage. The regulator contains two switchable silicon controlled rectifier (thyristor) bridges, either of which will supply rated alternator-exciter excitation. Two independent modes of regulation are provided:

- (a) ac automatic regulator to maintain desired generator stator terminal voltage, and
- (b) dc regulator to maintain constant generator field voltage.

The ac and dc regulators are also switchable. Either regulator can operate with either parallel bridge.

The turbine-generator system is described in Table 3.10.

3.3.2.4 Water Storage and Makeup Subsystem

The river water is assumed to be sufficiently clean to permit the use of a simple makeup water treatment plant. Water is taken from the service-water main into the clarifier tank, into which various chemicals are introduced by means of chemical feed pumps. For pretreatment in the clarifier, lime and a coagulant (alum) are fed simultaneously for cold lime softening and removal of turbidity. The water from the clarifier overflows to the clear well. Three 150-gpm-

TABLE 3.10

TURBINE-GENERATOR DESCRIPTION

TURBINE

TC-6F, 33-1/2 in. LSB
Rated Output - 1,055,600 kW
Initial Steam - 2400 psia, 950° F
Back Pressure - 1.5 in. Hg Abs.
Extraction - to 7 Stages of Feedwater Heaters, BFP Turbines
Reheater Steam - 528.4 psia, 950° F
3600 rpm

TURBINE ACCESSORIES

Oil Pumping and Cooling System including reservoir
Protective Valve System including main stop valves, control valves
Seal Steam System
Exhaust Spray System
Turning Gear
Instrumentation and Protective Devices
Piping between main stop valves and turbine inlet
Piping between Reheat stop valves and turbine inlet

ELECTRO-HYDRAULIC CONTROL SYSTEM

Fluid Coolers and Pumps
Two Positive Displacement Pumps and two 150 - hp ac Motors
Fluid Reservoir
Control Cabinet
Piping, Valves, Strainers, Switches

GENERATOR

Direct Coupled, 3-Phase, 60 Cycle, 22,000 Volts
3600 rpm, 0.90 pF, 0.58 SCR
32,000 Amperes, Wye Connection, 2 Winding
1100 MVA
Hydrogen and Water Cooled
60 psi-H₂

GENERATOR ACCESSORIES

Hydrogen Cooler System
External Hydrogen System
Seal Oil System
Stator Cooling System
External Hydrogen Control System
Grounding Transformer

TABLE 3.10 (Cont'd)

ALTERREX EXCITER (Silicon Diode Rectifier)

3600 rpm, 500 Volts, 3200 kW
 Air-Water Heat Exchanger System
 Driver Coupling and Shaft
 Operating Instrumentation
 Metal Enclosure

capacity transfer pumps (one operating, one standby, and one backwash) take suction from the clear well and deliver the water through three pressure filters to the demineralizer trains.

The makeup water system is a two-train demineralizer system, and each train includes a cation exchanger, an anion exchanger, and a mixed-bed polishing exchanger. The system will process filtered water in sufficient quantity to provide makeup water for normal plant operation. In order to provide such large quantities of water, one cation-anion regeneration is required approximately every two days. One of the two trains provides full flow capability during periods of regeneration. Since no treated water storage tank is provided, a backup system is required in case of system breakdown or equipment malfunction.

The makeup water system components are described in Table 3.11.

The water treatment plant contains acid and caustic regeneration systems including tanks, pumps, and interconnecting piping and valving. The entire plant is fully automatic and pushbutton-operated.

3.3.2.5 Condensate Demineralizer Subsystem

Condensate from the main steam condenser passes through a full-flow demin-

eralizer system after leaving the gland seal condenser and before entering the feedwater heater train. These polishing demineralizers are of the "Powdex" type, which discards depleted resin upon backwash. The system includes six ion exchanger vessels, five of which accommodate full flow, with the sixth on standby.

The Powdex process employs a thin layer of powered ion-exchange resin applied as a precoat on the outer circumference of vertical, cylindrical elements which have pore openings sufficiently fine to retain the precoat material. The elements are attached to a tube sheet fixed within a pressure vessel. In mechanical design and operation, the Powdex unit is nearly identical to a conventional precoat filter. Spent resin is displaced from the elements by a reverse flow of air and water and flushed from the pressure vessels to the disposal area. The spent resin is not reclaimed. A fresh precoat is applied by mixing Powdex resin and dispersant in an external vessel and circulating the prepared slurry through the Powdex vessel, thereby depositing the Powdex resin on the filter elements.

The feedwater purity requirements are summarized in Table 3.12.

3.3.2.6 Service and Circulating Water Subsystems

The service water system provides cooling water to power plant air condition-

TABLE 3.11

DESCRIPTION OF PLANT MAKEUP WATER SYSTEM COMPONENTS

Component	Quantity	Description
Lime Feeder	1	Unit
Transfer Pump and Motor	1	Unit
Coagulant Feeder	1	Unit
Transfer Pump and Motor	1	Unit
Coagulant Aid Feeder	1	Unit
Feed Pump	1	Metering type
Spare Feeder	1	Unit
Transfer Pump and Motor	1	Unit
Clarifier	1	14 ft diameter × 12 ft high, 150 gpm protective - lined steel tank with agitator incl. motor drive
Clearwell	1	9 ft diameter × 12 ft high 5,700 gallon steel tank with protective lining
Transfer Backwash and Common Pumps	3 3	150 gpm ea. 30 psi 5 hp motors
Pressure Filters	3	4 ft 6 in. diameter × 5 ft high (8 ft overall) 34 gpm ea. normal 50 gpm ea. maximum Anthracite filter media Push-button automatic
Mixed Bed Ion Exchanger Train	6	3 ft 6 in. diameter × 12 ft high vessels
Caustic Heating Tank	1	Unit carbon steel
Caustic Tank	1	5,000 gal carbon steel
Acid Tank	1	5,000 gal
Neutralizing Tank	1	7,000 gal

TABLE 3.12

FEEDWATER PURITY REQUIREMENT - MAXIMUM IMPURITY

Total Dissolved Solids	0.05 ppm
Silica (as SiO ₂)	0.02
Iron (as Fe)	0.01
Copper (as Cu)	0.01
Colloidal Solids	0.00
Hardness	0.00
Organic	0.00
Free Caustic	0.00
Dissolved Oxygen	0.007
Carbon Dioxide	0.005
pH (controlled by excess hydrazine - N ₂ H ₂)	9.0 to 9.5

ing equipment, air compressors, nitrogen coolers, and decay heat condenser. Three service water pumps are provided, each having a 5500-gpm capacity with 125-foot total dynamic head. Two of the three pumps are sufficient for normal operation. The pumps are of the multistage vertical type, coupled directly with their motor drives. To assure an uninterrupted supply of cooling water, two full-capacity distribution headers are joined into a continuous loop. Sectionalizing valves facilitate maintenance of various portions of the loop without impairing their operation. Service water is supplied through a distribution system, as shown on Figure 3.13.

The main condenser is cooled by raw river water supplied by two half-capacity, 160,000-gpm circulating water pumps. These pumps are accommodated in a reinforced concrete intake structure located at the river's edge. The structure is partially submerged below the low-water level, assuring the proper pump submergence required for cavitation-free operation, even under extreme

low-water-level conditions. The structure includes four intake bays complete with bar racks, stop logs, traveling screens, and two 1200-gpm screen wash pumps and associated valves and piping. Each of the four traveling screens is made of monel, 11 feet wide by 33 feet long with 3/8-inch openings.

The circulating water pumps discharge into two 84-inch diameter pipes with protective lining, running below grade. These pipes carry circulating water approximately 200 feet to the main condenser. Circulating water is discharged from the main condenser through two 84-inch-diameter pipes. Each discharge pipe is equipped with a motor-operated butterfly valve, which facilitates filling the condenser tubes at startup and keeping them flooded at loss of circulating water pump power.

The discharge pipes run below grade from a point under the turbine building for approximately 1000 feet to the discharge structure located downstream at the river's edge. The discharge structure is designed

to prevent air from entering the pipes and breaking the siphon even under lowest water level conditions.

The service and circulating water system components are described in Table 3.13.

3.3.3 Principles of Operation

Prior to startup, the feedwater is circulated through the condensate demineralizers to assure its purity. An auxiliary boiler provides steam to operate the air ejectors and gland seals, and to heat the feedwater. The steam generator and the turbine are bypassed.

When the reactor is at very low power, purified feedwater is circulated through the evaporator and a separator to the condenser. Saturated steam from the separator is used to preheat the turbines and the steam piping. During the startup procedure, the saturation temperature of steam from the separator is held below the temperature of the superheater to prevent condensation in the superheater tubes.

As the reactor power increases and approaches the low power end of the normal operating range, the temperature and pressure of the steam and water mixture in the separator is increased. When the pressure in the separator reaches 800 psia, steam is admitted directly into the superheater from the evaporator through a throttling valve. Flow through the separator is stopped. Steam from the superheater bypasses the turbines and is desuperheated before it enters the condenser.

When the reactor power enters the normal operating range, steam is admitted to the turbines which are then carefully brought up to full speed.

During normal operation, the pressure of steam entering the turbine and the turbine

speed are held constant. The steam temperature decreases slightly as the plant power is increased. The rate of change of power is determined by the plant control system (Section 3.13).

3.3.4 Safety Precautions

The following are measures which must be taken to assure the safe operation of the steam, feedwater, and makeup water system:

1. The purity of the feedwater must be continuously monitored to protect the stainless steel components,
2. Materials samples which are placed in the system must be periodically removed and examined for corrosion and deposition,
3. The performance of the turbine overspeed trips and of the relief valves must be periodically checked,
4. All rotating equipment must be periodically checked for excessive vibrations and abnormal temperatures, and
5. Samples of the turbine lubricating oil must be periodically examined for water in order to detect leaking seals.

3.3.5 Maintenance

Adequate space has been provided in the turbine building to allow the replacement of any of the system components. A machine shop and warehouse are provided for on-site repairs.

Because of the absence of radiation from the steam, feedwater, and makeup water system, the maintenance procedures are not different from those employed at a fossil fueled power plant.

FIGURE 3.13 SERVICE WATER AND CIRCULATING WATER SYSTEM (SIMILAR TO BECHTEL DRAWING M-13)

TABLE 3.13

DESCRIPTION OF SERVICE AND CIRCULATING WATER
SYSTEM COMPONENTS

Component	Quantity	Description
<u>Circulating Water System</u>		
Circulating Water Pumps	2	1/2-capacity vertical mixed flow Centrifugal 160,000 gpm, 33 ft TDH
	2	1600 hp motors, 13,800 bolts
Travelling Screen	4	11 ft wide 33 ft long (20 ft submerged) screen material monel, 3/8 in. openings, water velocity = 1.5 fps
	4	5 hp motors
Screen Wash Pumps	2	1200 gpm, 200 ft TDH
	2	75 hp motors, 480 volts, three-phase, 60 cycle
Chlorine Injection System	1	Complete unit including chlorinator pumps, chlorine gas control valve, chlorine dispenser, chlorine solution diffusers
<u>Service Water System</u>		
Service Water Pumps	3	1/2-capacity pumps, motor-driven, 5,500 gpm ea., 125 ft TDH vertical centrifugal type
	3	200 hp motors, 480 volts, three-phase, 60 cycle

3.4 AUXILIARY COOLING SYSTEM (W.R. Gee)

3.4.1 Introduction

3.4.1.1 System Functions

The following are the two major functions of the auxiliary cooling system:

1. After reactor shutdown, lower the temperature of the primary sodium to allow reactor refueling, and
2. After a loss of power to the site, assure adequate cooling of the reactor core.

3.4.1.2 Summary Description

Cooling of the primary sodium after a normal shutdown is done by the components of the startup system (Section 3.13). Feedwater is circulated through the evaporator, the steam separator, and back to the condenser. The superheater and the turbines are bypassed.

After a loss of power to the site, followed by a reactor scram, heat is rejected from the steam through a high pressure condenser. Power is provided from the emergency power system to low power motors at the following locations:

1. Primary sodium pumps,
2. Secondary sodium pumps,
3. High pressure condenser hotwell, and
4. Service water circulating pump.

3.4.1.3 Design Requirements

The auxiliary cooling system must remove heat from the reactor core during both normal and emergency shutdowns. The following are specific requirements of the system:

1. During a normal shutdown, the temperature of the primary sodium must be reduced to 400° F within nine hours,
2. The peak cladding temperature in the core must not exceed 1300° F, and
3. Stresses due to thermal gradients in any of the components of the main heat transfer systems must not exceed the local allowable values.

3.4.2 Detailed Description

The auxiliary cooling system normally utilizes the components of the main heat transfer systems:

1. The primary heat transfer system (Section 3.1),
2. The secondary heat transfer system, (Section 3.2), and
3. The steam, feedwater, and makeup water system (Section 3.3).

During a normal shutdown, the temperatures throughout the plant will be decreased as rapidly as possible without developing excessive stresses. Both radioactive decay energy and stored energy are rejected through the main condenser.

However, if no outside power is available to the site at the time of shutdown, the electrical power required for emergency cooling will be minimized by rejecting only the radioactive decay energy. No attempt will be made to lower the temperature of the sodium in the primary and secondary heat transfer systems during the loss of power emergency.

To protect the evaporator from large thermal gradients, heat is rejected through a high pressure condenser during the emergency shutdown. The temperature of condensate from this condenser is not greatly different from the sodium outlet temperature of the evaporator. Because of its low design pressure, the main condenser produces condensate which is too cold for direct return to the evaporator.

Two full-capacity condensate pumps are provided to guarantee circulation of the high pressure condensate through the evaporator.

The high pressure condenser is cooled by water from the service water system (Section 3.3). Any one of the three half-capacity service water pumps will supply adequate flow. Power for the condensate pumps and the service water pumps is provided by the emergency power system (Section 3.10).

The following are the characteristics of the high pressure condenser and its condensate pumps:

Decay Heat Condensate Pumps	2 units, 350 gpm, 850 psia discharge, horizontal split case, single stage.
Decay Heat Condensate Pump Drives	2 units, 250 hp, 480 volts, 3 phase, 60 cps, 1760 rpm, squirrel cage motors.
Decay Heat Condenser	1 unit, 350 gpm, 1000 psig design pressure, 170×10^6 Btu/h, stainless steel, 4×10^3 ft ² heat transfer surface.

3.4.3 Principles of Operation

The auxiliary cooling system has two purposes:

1. During normal shutdown cooling it reduces the maximum bulk temperature of the primary sodium to 400° F in nine hours after reactor shutdown to allow refueling of the reactor core, and
2. During emergency cooling it assures that no damage occurs to the reactor core or to the plant systems after a total loss of outside power to the reactor site. The scram system is presumed to operate.

The reactor power level after shutdown is assumed to decrease with time as shown in Figure 3.14. The normal operating temperature of most of the 3.0×10^6 pounds of

primary sodium is 810° F. To reduce the temperature of this sodium to 400° F in nine hours requires an average heat rejection rate of 12 MW (one-half per cent of full power). The radioactive decay energy and stored energy rejection requirements are additive.

The consequence of not rejecting the radioactive decay energy is summarized in Figure 3.15. If the primary sodium pumps continue to operate at reduced flow, the bulk temperature of the primary sodium will slightly exceed 1000° F at the end of the first hour after reactor shutdown.

After a normal reactor shutdown, the radioactive decay energy and the stored energy is rejected through the normal heat transfer systems to the main condenser cooling water. After an emergency shutdown, radioactive decay energy is rejected to the cooling water through a special high pressure condenser. Circulation of sodium, of condensate, and of cooling water is assured by electric motors powered from the Emergency Power System (Section 3.10.8).

3.4.3.1 Normal Reactor Shutdown Cooling

Following a normal reactor shutdown, energy is rejected through the normal heat transfer systems to the river water using one secondary loop. The normal shutdown procedure is nearly the reverse of the start-up procedure described in Section 3.13. Reactor scram is not utilized for normal shutdown because of the possibly severe thermal shocks which result.

The primary sodium pumps are rotated at 20% of their full speed by their pony motors during the shutdown. The bulk primary sodium temperature is 400° F and the temperature rise through the core is less than 12° F.

FIGURE 3.14 CORE POWER AFTER SCRAM FROM EXTENDED FULL POWER OPERATION

FIGURE 3.15 PRIMARY SODIUM BULK TEMPERATURE INCREASE AFTER SCRAM FROM FULL POWER OPERATION

The secondary sodium pump of the one selected secondary sodium loop is operated at 50% of its full speed by its main drive motor. The cold leg temperature of that secondary sodium loop will be 350°F and the temperature rise through the intermediate heat exchanger is less than 20°F. The pumps of the inactive secondary loops are rotated at 20% of full speed by their pony motors.

The temperature of primary sodium from the active intermediate heat exchanger is 60°F colder than that from the inactive ones. However, the long transit time (11 minutes) through the primary system at low flow allows adequate mixing. Steam flows from the two evaporators of the selected loop through the startup steam separator and the desuperheater to the main condenser. Feedwater is supplied to the two evaporators at 90 psia by the condensate pumps and by the startup feedwater pump. The flow of feedwater is controlled to maintain the bulk temperature of the primary sodium at 400°F.

One of the two parallel circulating water loops is operated to provide cooling water to the main condenser. The main condenser is arranged so that circulating water from one loop flows through only half of the condenser tubes (divided water box).

3.4.3.2 Emergency Cooling

Following a loss of power to the plant site and a reactor scram from full power, energy is rejected to the river water through a special high pressure condenser. The primary and secondary sodium pumps coast down to 20% of full speed and are maintained at that speed by power from the emergency power system to the pump pony motors. Both the condensate pumps for the high pressure condenser and the service water pump are also operated by emergency power. The

high pressure condenser is cooled by water from the service water system.

The high pressure condenser and its condensate pump are designed for a maximum operating pressure of 800 psia. The evaporator pressure decreases to this value as the feedwater pumps coast to a halt and as steam is bypassed through the desuperheater to the main condenser (or vented to the atmosphere). After the pressure in the evaporators has been reduced to 800 psia from its full power value of 2650 psia, the evaporator throttle valves are closed and the evaporator extraction valves (plant startup, Section 3.13) are opened to allow flow from the evaporators through the startup steam separator to the high pressure condenser. The subcooled condensate is returned to evaporator inlet at 500°F by one of the two parallel condensate pumps.

Cooling water is supplied to the high pressure condenser at atmospheric pressure and at ambient temperature. The thermal stresses resulting from the temperature difference between the cold feedwater and the hot condensate are minimized by providing a region of stagnant subcooled condensate on the high pressure side of the condenser. However, these stresses are quite high and may require replacement of the condenser after the emergency.

During the first minutes following the reactor scram and the loss of power, the plant components are exposed to thermal transients. Because of inertia in the pumps and drives, the core power decreases far more rapidly during the first seconds than does the sodium flow rate. Consequently, the core outlet temperature drops from 1150°F to less than 865°F in the first second. The outlet temperature then rises smoothly to 930°F after 11 seconds. From this point the pumps rotate at 20% of full speed and the core outlet temperature decreases slowly.

The temperature of secondary sodium leaving the intermediate heat exchanger also decreases rapidly from its full power value of more than 1071° F to less than 713° F. The colder secondary sodium requires nearly 10 seconds to reach the steam generating components.

The main turbine stop valves are tripped simultaneously with the reactor scram. The boiler feed pumps coast to a stop. Steam flows through the bypass line to the desuperheater and the main condenser to rapidly depressurize the steam system. The steam dump valve in the turbine bypass line is spring loaded to close when the pressure drops below 800 psia. This action results in maintaining the temperature on the steam side of the evaporator at more than 500° F.

At this point steam flow is established through the high pressure condenser. The flow of cooling water through the condenser is regulated to maintain a constant condensate level in the high pressure condenser. As the reactor power continues to decrease, the flow of cooling water will decrease.

3.4.4. Safety Precautions

The following are measures which must be taken to assure the safe operation of the auxiliary cooling system:

1. The temperature distribution on the primary sodium tank must be continuously monitored when two secondary loops are inactive, to prevent excessive stresses,
2. The pony motors must be operated periodically to confirm their availability,
3. The emergency power system must be checked periodically to confirm its availability (Section 3.10),
4. After an emergency shutdown with loss of outside power, the high pressure condenser must be examined for evidence of excessive stresses, and replaced if necessary, and
5. The automatic control valves and pumps must be periodically checked to confirm their availability.

3.4.5 Maintenance

All of the components of the auxiliary cooling system are easily accessible for repair or replacement. However, the reactor must be shut down before any components are disturbed.

During the replacement of a sodium pump pony motor, one of the other pumps will be operated by its main drive to provide adequate reactor cooling.

Prior to the replacement of the high pressure condenser or one of its condensate pumps, the sodium systems must be cooled to 400° F. Heat can then be rejected through the main condenser without overstressing the evaporator.

3.5 SODIUM SERVICE SYSTEM (W. R. Gee)

3.5.1 Introduction

3.5.1.1 System Functions

The sodium service system provides the functions of receiving, storing, transferring, and purifying the sodium in the primary and secondary sodium systems, and in the irradiated fuel storage tank (See Figure 3.16). Components of the sodium service system are duplicated where necessary to avoid the mixing of radioactive and non-radioactive sodium.

3.5.1.2 Summary Description

Sodium for the initial charge and for makeup will arrive at the site in railroad cars. Each car will be heated to melt the sodium. The sodium will be withdrawn by pressurizing; it will be filtered and transferred to storage tanks. It is possible to circulate the stored sodium through cold traps.

Sodium may be transferred from the storage tanks to the primary and secondary sodium systems, and to the irradiated fuel storage tank. The sodium may be returned to the storage tanks. The sodium may be circulated through cold traps from the main systems at a small fraction of the main system flow rate.

3.5.1.3 Design Requirements

Table 3:14 summarizes the design bases for the sodium service system.

3.5.2 Detailed Description

3.5.2.1 Cold Traps

Cold traps are utilized to remove dissolved impurities from the sodium while it is either in storage, in the primary heat transfer system, in the secondary heat transfer system, or in the irradiated fuel storage tank. The impurities are expected to be primarily the oxides and carbides of sodium.

Before entering the cold traps, the contaminated sodium passes through a regenerative heat exchanger. This economizer lowers the temperature of the contaminated sodium to a value slightly above the indicated plugging temperature. The contaminated sodium is then cooled slowly to a temperature slightly below the indicated plugging temperature as it passes through a bed of stainless steel mesh in the cold trap. Con-

taminants are removed through crystal growth of precipitates in the mesh.

Table 3.15 presents the characteristics of the cold traps for the radioactive sodium (primary heat transfer system, irradiated fuel storage tank) and for the non-radioactive sodium (secondary heat transfer system). The numbers shown on the table apply when the traps are being used for "polishing," i. e., reducing oxide concentration from 10 ppm to approximately 5 ppm. At higher oxide levels in the bulk sodium, the outlet temperature of the trap must be correspondingly higher (within 50° F of the indicated plugging temperature).

The cold traps which handle non-radioactive sodium are shown to be oversized by more than a factor of six. This allows utilizing the same design for both the radioactive and the non-radioactive sodium cold traps. Further, the secondary sodium is more subject to continuous contamination from small leaks of high pressure water and steam. The reserve capacity is desirable to minimize the frequency of cold trap replacement.

The design of the cold trays is shown in Figure 3.17. Sodium makes two passes through the mesh bed of the cold trap. It is cooled on a downward pass through the outer annulus and heated slightly on an upward pass through the center.

The cold traps are cooled by forced convection of nitrogen. Nitrogen is received from the nitrogen system at 90° F and is heated to 150° F. Nitrogen pressure drop through the cold traps is 20 inches of water.

The radioactive sodium cold traps and economizers are located in a nitrogen-filled vault inside the reactor building. The vault is shielded with concrete to attenuate

FIGURE 3.16 SODIUM SERVICE SYSTEM (SIMILAR TO GE DRAWING 174F127)

FIGURE 3.17 SODIUM COLD TRAP AND ECONOMIZER
(SIMILAR TO GE DRAWING 106D3526)

TABLE 3.14

DESIGN BASES FOR THE SODIUM SERVICE SYSTEM

Normal Oxide Solubility, ppm of Oxygen	5
Normal Sodium Plugging Temperature, ° F	230
Maximum Sodium Temperature, ° F	1000
Number of System Volumes for One Cold Trapping Run	3
Time Allowed for One Cold Trapping Run, h	168
Sodium Residence, Time In Cold Traps, min	5
Sodium Temperature Drop in Cold Traps	50
Number of Cold Traps for Radioactive Sodium	4
Number of Cold Traps for Non-Radioactive Sodium	6
Mass of Radioactive Sodium, pounds	3.0×10^6
Mass of Non-Radioactive Sodium, pounds	0.7×10^6
Oxide Resulting from In-leakage of One Refueling Cell	4.03×10^3
Volume of Air, pounds	
Oxide Resulting From In-leakage of One Evaporator	28.0×10^3
Volume of Water, pounds	

TABLE 3.15

CHARACTERISTICS OF THE SODIUM COLD TRAPS

	<u>Radioactive</u>	<u>Non-Radioactive</u>
Total Number (in parallel)	4	6
Total Required Clean-up Flow Rate, lbs/h	5.4×10^4	1.3×10^4
Design Clean-up Flow Rate Per Trap, lbs/h	1.4×10^4	1.4×10^4
Design Clean-up Flow Rate Per Trap, gal/min	32	32
Maximum Total Cooling Load, Btu/h	8.5×10^5	1.3×10^6
Required Volume Per Trap, cu ft	20	3.1
Design Volume Per Trap, cu ft	20	20
Design Volume Per Trap, gal	150	150
Depth of Packing, inches	35	35
Diameter of Packing, inches	35	35
Maximum Oxide Capacity of Each Trap, lbs	560	560
Design Sodium Pressure, psig	20	60
Design Sodium Temperature, ° F	1000	1000
Material of Construction	304 SS	304 SS

TABLE 3.16

CHARACTERISTICS OF THE SODIUM COLD TRAP ECONOMIZERS

	<u>Radioactive</u>	<u>Non-Radioactive</u>
Total Number	4	6
Number Per Cold Trap	1	1
Material of Construction	304 SS	304 SS
Contaminated Sodium Normal Inlet Temperature, ° F	850	680
Contaminated Sodium Minimum Outlet Temperature, ° F	280	280
Purified Sodium Minimum Inlet Temperature, ° F	230	230
Purified Sodium Normal Outlet Temperature, ° F	800	630
Mean Temperature Difference, ° F	50	50
Sodium Flow Rate, (each), lbs/h	1.4×10^4	1.4×10^4
Heat Duty, Btu/h	2.5×10^6	1.8×10^6
Number of Tubes per Economizer	19	7
Tube Inside Diameter, inches	2.0	2.0
Pitch-to-Diameter Ratio (Triangular Pitch)	1.5	1.5
Tube Wall Thickness, inches	0.05	0.05
Shell Diameter (Schedule #40S Pipe), inches	16	10
Overall Heat Transfer Coefficient, Btu/h ft ² ° F	1000	1000
Active Area per Economizer, ft ²	100	36
Tube Active Length, ft	9.6	9.4
Design Sodium Pressure, psig	20	60
Design Sodium Temperature, ° F	1000	1000

the nuclear radiation from contaminants in the radioactive sodium. The vault is lined with steel sheets to protect the concrete from molten sodium during an inadvertant leak.

The non-radioactive sodium cold traps and economizers are located in a similar nitrogen filled vault outside of the reactor building. The requirement for nuclear radiation shielding is greatly decreased.

3.5.2.2 Cold Trap Economizers

Regenerative heat exchangers are utilized to minimize the heat loss in the sodium cold traps. One economizer is arranged in series with each cold trap. It is not necessary to replace the economizer when replacing a cold trap. Table 3.16 presents the characteristics of the cold trap economizers.

The economizer is a parallel counter-flow shell-and-tube heat exchanger. The tubes are welded at one end to a fixed tube sheet and at the other end to a sliding tube sheet. Small leakage from the shell side to the tube side occurs. (See Figure 3.17).

High impurity sodium flows on the shell side to minimize the likelihood of plugging. Purified sodium flows through the tubes. The economizer is vertically mounted with the high temperature regions uppermost to enhance flow stability.

3.5.2.3 Sodium Service System Expansion Tanks

Expansion tanks are provided at the high points in the sodium service system to facilitate draining the system and to accommodate sodium expansion in the system. Table 3.17 summarizes the characteristics of the service system expansion tanks.

3.5.2.4 Sodium Service System Pumps

Sodium service system pumps are utilized to transfer sodium from storage to the main systems and to circulate sodium through the cold traps. Linear induction electromagnetic pumps were chosen for this application because of their wide range of flow control, and lack of shaft sealing problems.

Table 3.18 presents the characteristics of the sodium service system pumps. The maximum flow rate while filling the main systems was chosen arbitrarily to be that required to fill one secondary loop in eight hours. The higher developed head capability is required for the non-radioactive pump because of the large elevation differences in the secondary heat transfer system.

3.5.2.5 Plugging Indicators

Plugging indicators are used to measure the plugging temperature of the sodium being purified. The plugging indicator as-

sembly consists of an electromagnetic flow meter, an economizer, a nitrogen-cooled section, a temperature indicator, and a flow restriction with an electrical heater. The assembly is arranged so that approximately one gallon per minute of sodium flows through, in parallel to the flow through the cold traps.

Two plugging indicators are required. One is for radioactive sodium, and the other is for non-radioactive sodium.

To utilize the plugging indicator, continuous measurements of flow and temperature are made as the sodium passing through the unit is cooled slowly. When a temperature is reached which causes precipitation of impurities from the sodium, the precipitates will be trapped in the flow restriction. A sudden "break" in the sodium flow rate will be noted. The temperature at which the break occurs is assumed to be the system plugging temperature.

The plug is then redissolved using the electrical heater, and flow through the plugging indicator is re-established.

3.5.2.6 Sodium Receiving

Sodium will arrive at the plant site in railroad tank cars. Each car will contain 80,000 pounds of solidified sodium and an argon cover gas. The tank cars are equipped with oil channels welded to the exterior of the tank shell. Five-hundred gallons per minute of oil at 270° F will be circulated through the oil channels. Eight to 12 hours will be required to heat the sodium in the tank car to approximately 250° F. The required heat input is about 1.0×10^6 Btu/h.

The car will be pressurized with argon gas to force the molten sodium from the tank car, through a metallic filter, and into either the primary or the secondary sodium storage tanks.

TABLE 3.17

CHARACTERISTICS OF THE SODIUM SERVICE SYSTEMEXPANSION TANKS

	<u>Radioactive</u>	<u>Non-Radioactive</u>
Total Number	1	1
Cold Trap Volume, ft ³	80	120
Piping Volume, ft ³	2	4
Maximum Temperature Change, ° F	600	600
Maximum Volume Change, ft ³	7	11
Design Volume, ft ³	14	22
Design Volume, gal	105	165
Height, ft	2.6	3.0
Diameter, ft	2.6	3.0
Design Pressure, psig	15	15
Design Temperature, ° F	1000	1000
Material of Construction	304 SS	304 SS

TABLE 3.18

CHARACTERISTICS OF THE SODIUM SERVICE SYSTEM PUMPS

	<u>Radioactive</u>	<u>Non-Radioactive</u>
Total Number	1	1
Type	Electromagnetic Linear Induction	
Suction Pressure Required, ft of Na	15	15
Design Pressure, psig	20	20
Design Temperature, ° F	1000	1000
Material in Contact with Sodium	304 SS	304 SS
<u>Filling Main System:</u>		
Maximum Flow Rate, lbs/h	3×10^4	3×10^4
Maximum Flow Rate, gal/min	68	68
Maximum Developed Head, ft of Na	50	120
Maximum Developed Head, lb/in ²	19	46
Maximum Developed Horsepower	7.5	18.2
<u>Circulating Through Cold Traps:</u>		
Maximum Flow Rate, lbs/h	5.4×10^4	8.4×10^4
Maximum Flow Rate, gal/min	123	191
Maximum Developed Head, ft of Na	5.3	5.3
Maximum Developed Head, lbs/in ²	2	2
Maximum Developed Horsepower	1.5	2.2

3.5.2.7 Sodium Storage

Two large storage tanks are provided to store, respectively, radioactive and non-radioactive sodium. The radioactive sodium storage tank is located within the reactor building in a nitrogen filled vault. The non-radioactive sodium storage tank is located outside of the reactor building, below grade, in a nitrogen filled vault. Both tanks are constructed of carbon steel. The design temperature is 500°F and the design pressure is 15 psig.

The volume of the radioactive sodium storage tank (16,000 gallons) was chosen, arbitrarily, to be one-half that of the non-radioactive sodium storage tank. No clear criterion is available to choose the size of this tank. Overflow of primary sodium due to massive failure of an intermediate heat exchanger can be accommodated within the primary sodium tank. During shutdown, level changes of the sodium over the core may be achieved through cover-gas pressure control.

The volume of the non-radioactive sodium storage tank (32,000 gallons) was chosen to accommodate the full charge from one secondary sodium loop. Table 3.19 lists the volume of the loop components and the volume of the non-radioactive sodium storage tank.

3.5.2.8 Other Components

All piping for the sodium service system is standard gauge two-inch (nominal) stainless steel piping. Smaller pipe sizes would produce still reasonable sodium velocities. However, two inches was chosen as an arbitrary minimum to decrease the complexity of the system.

The sodium valves in the sodium service system are stainless steel vertical stem, angle shutoff valves. These valves are described in an Allis-Chalmers report of their AEC sponsored valve design study (Reference 8). The stem is sealed with a double bellows arrangement. When the valve is full open, the conical-shaped plug rests against a sleeve surrounding the inner bellows.

TABLE 3.19

SODIUM WETTED VOLUME OF ONE OF THREE SECONDARY SODIUM LOOPS

Steam Superheater (1)	193 ft ³
Steam Reheater (1)	302 ft ³
Evaporators (2)	1,140 ft ³
Piping (300 ft at 30.6 in. i.d.)	1,530 ft ³
Intermediate Heat Exchanger	700 ft ³
Total Volume	3,865 ft ³
Total Volume	29,000 gal
Design Storage Volume	32,000 gal

Electrical resistance heaters are used to preheat the piping and components of the sodium service system. A single line of heaters each with a stainless steel foil reflector is strapped to the bottom of the pipes before the insulation is added. The heaters require a variable-voltage power source.

3.5.3 Principles of Operation

All of the components of the sodium service system can be heated by electrical resistance heaters which are anchored to the outside metal surface, beneath the insulation. Before sodium is introduced, all of the system components and piping are filled with argon gas. After introduction of sodium, argon is used as a cover gas in the tanks and in the empty piping and components.

Molten sodium is forced to flow from railroad cars to the storage tanks by argon pressure. This transfer is done at a temperature which is only slightly greater than the melting point of pure sodium. Most of the impurities are expected to precipitate and either be left in the tank car or be trapped in the filter.

Sodium from the storage tanks can be circulated through the cold traps to remove impurities which may have been dissolved from the piping and the tank walls.

The vertical stem, angle, shut-off valves are kept in either the fully open or fully closed position. The flow rate is controlled by the electromagnetic pump, not the valves.

Sodium can be circulated from the primary tank through the cold traps. However, flow can be initiated only after the suction line has been filled by lowering its argon pressure relative to the primary cover gas. The inlet of the suction line is at a depth in the primary sodium which is a higher elevation than the top of the core.

3.5.4 Safety Precautions

The following are measures which must be taken to assure the safe operation of the sodium service system:

1. Before removing a cold trap, its temperature must be carefully determined to assure that all of the sodium inside is frozen,
2. During replacement of any component the area must be continuously monitored for evidence of a sodium fire,
3. The cold trap vaults must be continuously monitored for radioactive sodium spills,
4. The radiation level in the cold trap vaults must be continuously monitored and one or more of the traps must be replaced when the level becomes excessive, and
5. The nitrogen atmosphere of the vaults must be continuously monitored to assure that the oxygen and moisture levels are below the values which will support sodium burning.

3.5.5 Maintenance

The cold traps must be replaced either when they can no longer retain the impurity precipitates or when their radiation level approaches the allowable maximum. To remove a cold trap, the sodium within it must first be frozen. This is accomplished by closing the sodium valve and continuing the flow of nitrogen.

The cold trap is disconnected from the economizer and from the nitrogen system at the flanged joints. For the radioactive cold traps, this must be done remotely. The cold trap is then withdrawn into a shipping cask.

A new cold trap is then installed. No attempt is made to recover the spent cold trap.

Maintenance of other components of the system is also accomplished by this procedure of local freezing and component replacement.

3.6 STEAM GENERATOR PRESSURE RELIEF SYSTEM (W.R. Gee, Jr.)

The potential exists for developing large pressures on the shell side of the steam generators. If a tube should fail, high temperature steam will mix with the molten sodium on the shell side. The resulting sodium-water reaction releases energy (approximately 3,120 Btu per pound of steam reacted) and hydrogen gas (0.111 pounds of hydrogen per pound of steam reacted). Either the in-leakage of steam or the evolution of large quantities of high temperature hydrogen gas will pressurize the steam generator shell beyond its design pressure of 350 psig if this pressure is not relieved.

Rupture discs are supplied in each of the two cover-gas spaces of each steam generator unit to relieve this pressure. When the pressure in the cover gas exceeds the design rating of the rupture discs, the reaction products are allowed to flow into a long pipe. The pipe is normally filled with ambient pressure nitrogen gas. The pipe conducts the reaction products away from the steam generator area to a pool of water located at the base of a stack. The reaction products are released at the bottom of the water pool. Any gaseous products rise through the water, are cooled, and are released through the stack. Any entrained sodium reacts with water to produce sodium hydroxide and hydrogen gas. The sodium hydroxide remain dissolved in the water for later disposal.

In addition, hydrogen sensors and pressure sensors are provided in communi-

cation with each cover gas space. When the hydrogen concentration in one of the cover-gas spaces exceeds background an alarm signal is produced. The plant operators then must quickly depressurize the steam side of the loop which produced the alarm. The high hydrogen concentration implies a small steam leak into sodium.

When the pressure of the cover gas in a given steam generator unit exceeds its normal value by a pre-set margin, a signal is produced by the pressure sensors. This signal activates valves to isolate the one secondary heat transfer loop which contains the affected steam generator, and also causes a reactor shutdown (run-in of all shim rods). Quick closing valves stop feedwater to the loop. Relief valves vent steam from the loop to the atmosphere. Isolation valves prevent backflow of steam from other loops. The turbine is bypassed and flow is established through the decay heat removal system.

The sodium charge of the secondary loop is not drained during the incident. Its presence is expected to dilute the caustic reaction products and provide a heat sink for the energy of reaction.

Table 3.20 summarizes the design requirements for the steam generator pressure relief system. The maximum inflow rates were conservatively calculated for flow through an isentropic nozzle. The actual inflow through a long tube with friction will be significantly less.

Only one rupture disc of approximately ten inches diameter can be provided in the inner cover gas space of each steam generator unit (see Figure 3.9). However, more than one rupture disc can be provided in the outer cover gas space. Ten-inch pipes conduct reaction products away from the steam generator to a twenty-inch pipe. The twenty-inch line collects reaction product flow from all rupture discs and conducts it to the bot-

TABLE 3.20

DESIGN REQUIREMENTS FOR THE STEAM GENERATOR
PRESSURE RELIEF SYSTEM

Evaporator

Steam Tube Inside Diameter, inches	0.625
Steam/Water Temperature, ° F	676
Steam/Water Pressure, psia	2650
Maximum In-Flow Rate from One Tube Guillotine Failure:	
Steam, lb/sec	20.8
Water, lb/sec	36.2
Total Hydrogen Gas, lb/sec	6.3

Superheater

Steam Tube Inside Diameter, inches	0.75
Steam Temperature, ° F	950
Steam Pressure, psia	2625
Maximum In-Flow Rate from One Tube Guillotine Failure:	
Dry Steam, lb/sec	24.5
Hydrogen Gas, lb/sec	2.7

Reheater

Steam Tube Inside Diameter, inches	1.50
Steam Temperature, ° F	950
Steam Pressure, psia	515
Maximum In-Flow Rate from One Tube Guillotine Failure:	
Dry Steam, lb/sec	19.6
Hydrogen Gas, lb/sec	2.2

tom of a pool of water at the base of the stack.

Table 3.21 summarizes the maximum relief rate from the evaporator through one 10-inch port. Two cases are presented: (1) venting to atmospheric pressure through an isentropic nozzle; and (2) venting to atmospheric pressure through a long ten inch pipe, with friction. The values shown for equivalent number of failed tubes imply a steady state relief rate (feedwater valves do not close and steam vent valves do not open).

The table shows that if the reaction rate of steam and sodium is slow, e.g., there is a tube break in the cover gas space above the sodium level, the steam from only a few failed tubes can be continuously vented. For this case reliable shut-off of feedwater is absolutely essential.

If the reaction rate of steam and sodium is rapid, the gas from far more tubes may be continuously vented. However, the rate of formation of sodium oxide would lead to rapid plugging of the affected secondary sodium loop. Again, the reliable shut-off of feedwater is essential.

TABLE 3.21

MAXIMUM CONTINUOUS RELIEF RATE OF EVAPORATOR
SHELL SIDE PRESSURE

Fraction of Maximum Steam/Sodium Reaction Rate	0.0		1.0	
Pressurizing Gas	Steam		Hydrogen	
Probable Gas Temperature, ° F	1000		2000	
Gas Pressure, psia	<u>150</u>	<u>300</u>	<u>150</u>	<u>300</u>
Vent to Atmosphere Through One 10-in. Nozzle:				
Flow, lb/sec	127	254	32.6	65.2
Equivalent No. of Failed Tubes	2.2	4.5	5.2	10.4
Vent to Atmosphere Through One 10-in. Pipe 20 ft Long:				
Flow, lb/sec	105	210	27	54
Equivalent No. of of Failed Tubes	1.8	3.7	4.3	8.6

3.7 IRRADIATED FUEL STORAGE COOLING SYSTEM (C.E. Boardman)

Two fuel storage tanks are provided in the reactor refueling cell. One tank is normally filled with argon gas and is used to store new fuel. The other tank is normally filled with molten sodium and is used to store irradiated fuel. Under unusual circumstances the first tank may be filled with sodium and used to store irradiated fuel.

The purpose of the irradiated fuel storage cooling system is to remove heat from the two storage tanks and to dissipate the heat to the atmosphere. Under normal

conditions the system must maintain the temperature of the irradiated fuel storage tank at 400° F. The tank is assumed to contain one third of the core fuel. The heat generation rate (specific power) is typical of the reactor three days after shutdown from extended full power operation.

Under loss of power conditions the system must maintain the temperature of the irradiated fuel storage tank below 800° F. This limit allows the utilization of carbon steel in the NaK loop.

When both tanks are being cooled the heat generation rate is typical of the re-

actor nine days after shutdown from full power operation.

The flow diagram for the system is shown on Figure 3.18. The required performance for the cooling system under normal conditions, with one-third of the core in storage, is as follows:

Heat Load	11.4×10^6 Btu/h
Sodium Temperature	400° F

Under abnormal conditions, with the full core loading in storage, the following performance is required:

Heat Load	34.2×10^6 Btu/h
Sodium Temperature	800° F

The irradiated fuel storage cooling system utilizes an independent NaK (eutectic composition) filled heat transfer loop. Heat is transferred from the sodium in the storage tanks to the NaK through coiled-tube heat exchangers submerged in the storage tanks. The NaK is circulated by one of two full-capacity electromagnetic pumps arranged in parallel. The NaK is piped vertically from the storage tank coils, at grade, to two half-capacity air cooled heat exchangers located on the roof of the reactor building. The NaK-to-air heat exchangers are cooled by forced convection of air. Each heat exchanger has two half-capacity blowers.

A NaK service system is provided to receive, to store, and to purify the NaK coolant.

During an emergency, one of the four fans at the NaK-to-air heat exchangers and one of the electromagnetic pumps are driven by the emergency power system (Section 3.10.3.8).

Table 3.22 contains a summary of the specifications for the irradiated fuel storage cooling system components.

3.8 INERT GAS SYSTEM (P.M. Tschamper)

3.8.1 Introduction

3.8.1.1 System Functions

The functions of the inert gas system are the following:

1. Minimize the likelihood of sodium fires by excluding oxygen and water vapor from areas where sodium spilling is most likely,
2. Provide cooling and thermal insulation for the sodium systems, and
3. Provide sodium level control in the vessels which contain sodium.

Because it is less expensive, nitrogen gas is used wherever possible in the inert gas systems. However, argon gas is used in those areas where the high temperature could lead to nitride embrittlement of the materials.

3.8.1.2 Summary Description

The inert gas system consists of two major subsystems: (1) the nitrogen subsystem, and (2) the argon subsystem. Argon is circulated to every location where it is necessary to maintain a free sodium surface and to the refueling cell. The pressure of the argon at these locations is regulated by the local level or pressure control instrumentation and valves.

The cover gas over the primary sodium is bled off at a continuous low rate. This gas passes through a nitrogen cooled heat exchanger where it is stripped of sodium vapor. The cooled and purified gas is then returned to the space above the gas baffles of the primary heat transfer system. This feature prevents the buildup of a frozen

FIGURE 3.18 IRRADIATED FUEL STORAGE COOLING SYSTEM (SIMILAR TO GE DRAWING 106D3628)

TABLE 3.22

IRRADIATED FUEL STORAGE COOLING SYSTEM
COMPONENT SPECIFICATIONS

Sodium-to-NaK Heat Exchangers

Type	Helical coiled tubes
Number (one spare)	2
Heat Transfer Surface	200 ft ²
Heat Transfer Rate	800 Btu/h ft ² ° F
Mean Temperature Difference	73° F
Tube o.d. (BWG 18)	1 inch
Material	304 stainless steel

Electromagnetic Pumps

Type	Linear Induction
Number (one spare)	2
Design Flow	680 gpm
Developed Head	15 psi
Suction Pressure	10 psi
Material	304 stainless steel

NaK-to-Air Heat Exchangers

Type	Finned Tube, cross flow
Number	2
Heat Transfer Surface	5200 ft ²
Heat Transfer Rate	7 Btu/h ft ² ° F
Mean Temperature Reference	157° F
Tube o.d. (BWG 15)	1 inch
Material	carbon steel
Number of Blowers	2

NaK Expansion Tank

Number	1
Capacity	120 gal
Material	carbon steel

NaK Transfer Tank

Number	1
Capacity	120 gal
Material	carbon steel

TABLE 3.22 (Cont'd)

NaK Drain Tank

Number	1
Capacity	1000 gal
Material	carbon steel

NaK Plugging Indicator

Number	1
Flow Rate	1 gpm
Material	304 stainless steel

NaK Cold Trap (Air Cooled)

Number	1
Oxide capacity	200 pounds
Flow rate	10 gpm
NaK temperature drop	50° F
Material	carbon steel

sodium layer on the cool surfaces of the primary containment vessel.

Nitrogen is circulated to the steel-lined concrete vaults which contain the sodium handling equipment. In addition it is used to cool the pumps and cold traps of the sodium service system, the primary containment vessel, and the primary cover gas. Heat is rejected from the nitrogen through a freon cooled heat exchanger.

3.8.1.3 Design Requirements

The inert gas system is required to exclude water vapor and oxygen from contact with sodium and with sodium handling equipment. The following are specific design requirements:

1. A continuous flow of cooled and purified argon must be provided to the cover gas space of the primary heat transfer system,

2. A continuous flow of cooled and purified nitrogen must be provided to the sodium service system, to the primary containment vessel, and to the argon coolers,
3. A continuous flow of cooled and purified nitrogen must be provided to the steel-lined vaults which contain sodium handling equipment, and
4. A reservoir of high pressure argon must be available to control the sodium level in the sodium handling equipment.

3.8.2 Detailed Description

3.8.2.1 Nitrogen Subsystem

In the reactor building, equipment which contains molten sodium is located in concrete cells which are lined with carbon steel sheets. The cells are sealed so that

their leakage rate is less than one percent of their total volume per day when their internal pressure is 10 psig. The equipment cells are filled with nitrogen gas. The gas pressure within the cells is maintained within the range from 1 to 5 inches of water greater than the reactor building air pressure. The pressure prevents the inleakage of air. The gas temperature within the cells is maintained below 150° F.

The sodium service system (Section 3.5) requires, in addition, nitrogen gas flow to cool the cold traps, the electromagnetic pumps, and the plugging indicators.

Nitrogen is also used to cool the argon which is the cover gas for the primary heat transfer system. Nitrogen and argon are forced through argon-to-nitrogen or nitrogen-to-freon heat exchangers to provide this cooling.

The heat loads and the gas flow requirements for the nitrogen system are summarized on Table 3.23. The maximum design heat load is 3.92×10^6 Btu/h (1.2 MWt). The maximum design flow rate is 2.13×10^5 lb/h (50,000 scfm).

A diagram of the nitrogen system is shown on Figure 3.19. Nitrogen is supplied at 90° F from a distribution header to five independent loops. The loops include the stations listed on Table 3.23 as follows:

I. Cover Gas Cooling Loop

1. Primary Sodium Tank Cover Gas Cooling
2. Reactor Vessel Cover Gas Cooling
3. Nitrogen Attenuator

II. Reactor Pot Cavity Loop

5. Primary Sodium Tank Cavity

III. Storage Tank Cooling Loop

6. Spent Fuel Storage Tank Cavity
7. New Fuel Storage Tank Cavity
8. Miscellaneous Storage Vault
9. Primary Sodium Storage Tank Vault

IV. Primary Sodium Service Loop

4. Cold Traps
13. Primary Service System E-M Pumps

V. Cell Nitrogen Loop

10. Cover Gas Cooling Cell
11. Radioactive Gas Cell
12. Primary Sodium Service System Cell
14. Reactor Building Pipeway
15. Mechanical Equipment Cell

The purity of the nitrogen in each of the five loops is continuously monitored. If either sodium or radioactive material leaks into the nitrogen system, the outlet flow from the affected loop is directed to the radioactive gas handling system (Section 3.9) and an alarm is produced. Further action must be taken by the operating personnel.

Flow from each of the five loops enters a return header at a temperature less than 150° F. Nitrogen then flows, in three parallel streams, through a damper, through a filter (with continuously moving filter material), through a nitrogen-to-freon heat exchanger, and through a centrifugal compressor. The nitrogen is cooled to 78° F in the heat exchangers. However, it leaves the compressors at a temperature of 90° F and a pressure of 20 inches of water (gauge). Nitrogen dryers are provided to maintain the dew point at a temperature 5° F less than the freon coil temperature.

TABLE 3. 23

HEAT LOADS AND COOLANT FLOW REQUIREMENTS FOR THE NITROGEN SYSTEM

<u>Station</u>	<u>Heat Duty (10^3 Btu/h)</u>		<u>Gas Flow (10^3 lb/h)</u>	
	<u>Required</u>	<u>Design</u>	<u>Required</u>	<u>Design</u>
1. Primary Sodium Tank Cover Gas Cooling*	300	450	2.9	4.4
2. Reactor Vessel Cover Gas Cooling*	225	337	2.2	3.3
3. Nitrogen Attenuator	95	143	3.5	5.4
4. Cold Traps	500	750	33.3	50.0
5. Primary Sodium Tank Cavity	550	825	36.7	55.0
6. Spent Fuel Storage Tank Cavity	287	430	19.2	28.7
7. New Fuel Storage Tank Cavity	287	430	19.2	28.7
8. Miscellaneous Storage Vault	30	45	2.0	3.0
9. Primary Sodium Storage Tank Vault	92	130	6.2	9.3
10. Cover Gas Cooling Cell	31.5	47.5	2.2	3.3
11. Radioactive Gas Cell	15	22.5	1.0	1.5
12. Primary Sodium Service System Cell	5	7.5	0.33	0.5
13. Primary Service System E-M Pump	66.5	100	4.5	6.7
14. Reactor Building Pipeway	8	12	0.54	0.8
15. Mechanical Equipment Cell	<u>120</u>	<u>180</u>	<u>6.7</u>	<u>12.0</u>
Totals	2612	3918	140.5	212.6

*Nitrogen temperature rises from 90° F to 500° F in the argon-to-nitrogen heat exchangers, and from 90° F to 150° F at all other stations.

Each compressor is rated at 50% of the total required system flow rate and each heat exchanger is rated at 50% of the required system heat duty. Flow through each compressor is controlled by operating the damper at its inlet.

Liquid nitrogen is stored at a station outside of the reactor building. The storage capacity corresponds to 10% of the mass of nitrogen in the nitrogen system (20,000 scf). Two half-capacity tanks, with independent valves and lines, are provided to assure adequate availability of the nitrogen supply. Each storage tank is arranged so that nitrogen can be extracted at a maximum rate of 100 scfm.

A diagram of the cover gas cooling loop of the nitrogen system is shown on Figure 3.20. Nitrogen enters the loop from the nitrogen system supply header with a temperature of 90° F and a gauge pressure of 20 inches of water. The supply stream is divided into two separate streams. One stream flows to the argon-to-nitrogen heat exchangers, and the other flows to two nitrogen-to-freon heat exchangers where the nitrogen temperature is reduced to 40° F.

Two heat exchangers are provided to cool the argon cover gas of the primary sodium tank (809° F), and two are provided to cool the argon cover gas of the reactor vessel (1150° F). Argon is cooled to 150° F in both sets of heat exchangers. Nitrogen enters both sets of heat exchangers, in parallel, at 90° F and is heated to 500° F.

The hot nitrogen then flows through two parallel nitrogen-to-water heat exchangers. The nitrogen is cooled to 225° F in these two heat exchangers. Two parallel centrifugal compressors force the 225° F nitrogen stream through an attemperator where it is mixed with the 40° F nitrogen stream. The mixture is then returned to

the nitrogen system with a temperature of 150° F and at atmospheric pressure.

The nitrogen-to-freon heat exchangers in the cover gas cooling loop also serve to dry the nitrogen inventory of the nitrogen system. The flow rate is such that a mass of nitrogen equivalent to the system inventory passes through the heat exchangers in one day. A tank is provided to collect the water condensed from the nitrogen (batch operation).

Table 3.24 summarizes the specifications for the nitrogen system equipment.

3.8.2.2 Argon Subsystem

Argon is used as a cover gas wherever a free surface of molten sodium is required. In these applications the pressure of the argon in the cover gas space is regulated by the local sodium level control systems. When a higher pressure is demanded, more gas is allowed to flow in from a higher pressure header. When a lower pressure is demanded, gas is exhausted to a lower pressure header.

Except for the components of the primary heat transfer system, the low pressure headers exhaust directly to either the radioactive gas handling system or the atmosphere. A gas recirculating subsystem is provided for the cover gas spaces of the primary heat transfer system components. This subsystem is required to cool parts of the primary heat transfer system components as well as to provide sodium level control.

In addition the refueling cell is filled with argon gas to minimize contamination of the primary heat transfer system cover gas during refueling.

Tables 3.25 and 3.26 summarize the flow and pressure requirements of the argon system. The low pressure header from the

TABLE 3.24

NITROGEN SYSTEM EQUIPMENT SPECIFICATIONSNitrogen Supply Tanks

Type	Cryogenic
Number	2
Capacity (10,000 scf)	14 ft ³
Operating Pressure	200 psia
Operating Temperature	-260° F
Maximum Extraction Rate	100 SCFM

Main Nitrogen Coolers

Type	Finned Tube, Nitrogen-to-Freon
Number	3
Heat Transfer Surface	3210 ft ²
Heat Load	1.31×10^6 Btu/h
Mean Temperature Difference	74° F
Nitrogen Flow	70,900 lb/h
Nitrogen Temperature, in/out	150/78° F

Nitrogen Dryers

Type	Finned Tube, Nitrogen-to-Freon, Water Separator
Number	2
Heat Load	32.8×10^3 Btu/h
Nitrogen Flow	2,700 lb/h
Nitrogen Temperature, in/out	90/40 ° F

Nitrogen Compressors

Type	Centrifugal Blowers
Number	3
Nitrogen Flow	70,900 lb/h
Suction Pressure	Atmospheric
Developed Head	20 in. Water
Drive Power	90 hp
Nitrogen Inlet Temperature	78 ° F

TABLE 3.24 (Continued)

NITROGEN SYSTEM EQUIPMENT SPECIFICATIONSNitrogen Compressors (Boosters)

Type	Centrifugal Blowers
Number	2
Nitrogen Flow	7,700 lb/h
Suction Pressure	Atmospheric
Developed Head	5 in. Water
Drive Power	3 hp
Nitrogen Inlet Temperature	225 °F

Nitrogen Filters

Type	Dry, Disposable, Travelling
Number	3
Nitrogen Flow	16,750 scfm

Primary Sodium Tank Cover Gas Coolers

Type	Finned Tube, Argon-to-Nitrogen
Number	2
Heat Load	300,000 Btu/h
Mean Temperature Difference	265 °F
Heat Transfer Surface	1,450 ft ²
Argon Flow (Shell Side)	3,400 lb/h
Nitrogen Flow (Tube Side)	2,900 lb/h
Argon Temperature, in/out	850/150 °F
Nitrogen Temperature, in/out	90/500 °F

Reactor Vessel Cover Gas Coolers

Type	Finned Tube, Argon-to-Nitrogen
Number	2
Heat Load	225,000 Btu/h
Mean Temperature Difference	315 °F
Heat Transfer Surface	1,000 ft ²

TABLE 3.24 (Continued)

NITROGEN SYSTEM EQUIPMENT SPECIFICATIONS

Argon Flow (Shell Side)	900 lb/h
Nitrogen Flow (Tube Side)	1, 100 lb/h
Argon Temperature, in/out	1150/150° F
Nitrogen Temperature, in/out	90/500° F

Intermediate Nitrogen Coolers

Type	Tube and Shell, Water Evaporator
Number	2
Heat Load	270, 000 Btu/h
Mean Temperature Difference	113° F
Heat Transfer Surface	250 ft ²
Nitrogen Flow	3, 850 lb/h
Nitrogen Temperature, in/out	500/225° F

non-radioactive sources exhausts to the atmosphere.

Figure 3.21 is a diagram of the argon system. Liquid argon for makeup is stored in insulated containers outside of the reactor building. A parallel supply line, with a valve which is normally locked closed, is provided for purging the refueling cell. Argon for purging the cell will be supplied directly from the supply trucks.

Argon flows from the storage area through a high pressure header to the various stations where it is used. The makeup argon is passed through a purification unit to assure that the gas is dry and free of oxygen. A special valving arrangement is provided where the argon supply header enters the reactor building to prevent backflow of contaminated gas in the event of a pressure loss.

A recirculating loop is shown in Figure 3.21 for cooling the refueling cell atmosphere. This loop is designed to transfer 130 kW from the argon of the refueling cell to the freon system.

A recirculating loop is shown on Figure 3.21 for controlling the pressure of the refueling cell atmosphere. The loop is required to maintain the cell pressure within 1/2 inch of water of the set-point, over a range from -6 to +6 inches of water.

A loop is shown in Figure 3.21 for maintaining the purity of the refueling cell atmosphere. This loop is required to maintain the oxygen content of the cell below 10 ppm. The maximum air leakage rate (at -6 inches of water) is 0.01 ft³/min. Oxygen is removed by bubbling the argon through a NaK filled container.

FIGURE 3.20 NITROGEN SYSTEM-COVER GAS COOLING LOOP (SIMILAR TO GE DRAWING 106D3617)

FIGURE 3.21 ARGON SYSTEM (SIMILAR TO GE DRAWING 277E239 SH. 1)

TABLE 3.25

ARGON SYSTEM FLOW AND PRESSURE REQUIREMENTS (RADIOACTIVE)

Service	No. of Units	Supply Line Pressure (psig)	Operating Pressure (psig)	Argon Max. Requirement/per Unit (scfm)	Total Max. Argon Demand (scfm)
Primary Sodium Storage Tank	1	50	0 - 1	50	50
Irradiated Fuel Storage Tank	2	50	0 - 1	30	60
Primary Sodium Service System Expansion Tank	1	50	0 - 15	20	20
NaK Storage Tank	1	50	0 - 1	15	15
NaK Expansion Tank	1	50	0 - 10	10	10
Refueling Cell Makeup	1	50	0 - 5	5	5
Refueling Cell Argon; Local Supply Header	1	50	0 - 50	50	50
Argon Locks	2	25	0 - 2	35	70
Material Transfer Ports	4	25	0 - 2	10	40
Reactor Vessel Cover Gas Sub-System	1	50	0 - 5	25	25
Reactor Pot/IHX Cover Gas Sub-System	1	50	0 - 3	25	25
Sodium Supply Station	1	50	5 - 7	30	30
NaK-Supply Station	1	50	0 - 5	5	5
					Total 405

TABLE 3.26

ARGON SYSTEM FLOW AND PRESSURE REQUIREMENTS (NON-RADIOACTIVE)

Equipment	No. of Units	Argon Supply Pressure (psig)	Operating Pressure (psig)	Argon Max. Supply Rate (scfm)	Max. Monthly Consumption/per Unit (scf)
Secondary Sodium Expansion Tank	3	50	0 to 1	25	2000
Secondary Service System Expansion Tank	1	100	5 to 65	25	2000
Evaporator: Inside Annulus Outside Annulus	6	50 100	~ 0 ~ 30	45 45	2000
Superheater: Inside Annulus Outside Annulus	3	50 100	~ 30 ~ 60	30 30	1200
Reheater: Inside Annulus Outside Annulus	3	50 100	~ 30 ~ 60	30 30	1500
Secondary Sodium Storage Tank	1	100	5 to 40	15	1500
Secondary Sodium Pump	3	100	5 to 60	25	3000

Figure 3.22 is a diagram of the subsystem which cools and regulates the argon flow to the cover gas spaces of the primary heat transfer system. Argon, at the local sodium temperature, is extracted from four areas in the primary heat transfer system:

1. Reactor vessel cover gas, 1150° F, 15 to 20 psia;
2. Intermediate heat exchanger cover gas, 1150° F, 15 to 18 psia;
3. Primary sodium pump cover gas, 809° F, 15 psia; and
4. Primary sodium tank cover gas, 809° F, 15 psia.

The argon is saturated with sodium vapor. Four argon-to-nitrogen heat exchangers are provided to cool the argon in the subsystem. Argon from the reactor vessel cover gas space flows through one of two heat exchangers which are in parallel. Argon from the other three areas is mixed and flows through one of the remaining two heat exchangers which are also in parallel. Valves are provided so that the argon side of the inactive heat exchanger may be connected in series upstream of the active heat exchanger. This raises the temperature of the inactive heat exchanger and melts the sodium which has condensed and frozen within it. The melted sodium is drained away and the heat exchanger is ready for use. This arrangement assures that the recirculated argon is always free of sodium vapor.

Argon flows from each set of heat exchangers to a set of two parallel centrifugal blowers. The blowers pressurize the argon to a value matching the demand signal from the sodium level control instrumentation in the reactor vessel and in the primary sodium tank, respectively. An additional set of blowers is provided to

match the required pressure in the cover gas space over the intermediate heat exchangers. The cooled and dried argon, at 150° F, enters each area at an elevation above the anti-convection baffles (Section 3.1).

The flow of primary sodium through any one of the intermediate heat exchangers may be stopped by raising the pressure of the cover gas to a value which slightly exceeds the pressure of the reactor vessel cover gas. To eliminate the possibility of an inadvertent flow stoppage, the normal pressure controlling blowers are selected so that they are incapable of providing sufficient pressure to stop the flow. An additional set of two parallel blowers is provided for stopping the primary sodium flow through one intermediate heat exchanger. Valves are provided so that the argon supply to the selected intermediate heat exchanger can be made to pass through the blowers.

Double walls are provided in the primary heat transfer system, wherever high temperature sodium must be separated from low temperature sodium (Section 3.1). The space between these double walls is filled with argon gas. At the slip joint provided in the pipe connecting the reactor vessel and the intermediate heat exchanger, argon leaks from the reactor vessel cover gas space to the primary sodium tank cover gas space.

A separate line is provided to return argon from the primary sodium tank cover gas supply header to the reactor vessel cover gas. Gas flow rate instrumentation controls the compressor in this line so that the makeup flow rate matches the leak rate of argon from the reactor vessel cover gas space.

Table 3.27 summarizes the specifications for the argon system equipment.

FIGURE 3.22 ARGON SYSTEM-REACTOR COVER GAS SUBSYSTEM (SIMILAR TO GE DRAWING 277E239 SH. 2)

TABLE 3.27

ARGON SYSTEM EQUIPMENT SPECIFICATIONSArgon Supply Tanks

Type	Cryogenic
Number	2
Capacity (80,000 scf)	102 ft ³
Operating Pressure	200 psia
Operating Temperature	-240 °F
Maximum Extraction Rate	500 scfm

Refueling Cell Pressure Control Tank

Number	1
Capacity	150 ft ³
Operating Temperature	70 to 150 °F
Operating Pressure	5 to 100 psig
Material	Carbon Steel

NaK Bubbler

	<u>Supply</u>	<u>Refueling Cell</u>
Number	1	1
Argon Flow Rate, scfm	1000	667
Argon Pressure Drop, psi	5	3
NaK Capacity, lbs	1,430	11,600
Material	Carbon Steel	

Argon Cooler (Refueling Cell Argon)

Cooling Fluid	Freon
Number	2
Heat Load, Btu/h	305,000
Active Surface, ft ²	1250
Mean Temperature Difference, °F	49
Argon Flow Rate, lb/h	62,000
Argon Inlet Temperature, °F	100
Argon Outlet Temperature, °F	70

Compressors (Refueling Cell Argon)

	<u>Circulation</u>	<u>Pressure Control</u>	<u>Purification</u>
Type	Centrifugal	Reciprocating	Centrifugal
Number	2	1	1
Argon Flow Rate, scfh	574,000	25	40,000
Inlet Temperature, °F	70	70	70

TABLE 3.27 (Cont'd)

ARGON SYSTEM EQUIPMENT SPECIFICATIONS

Suction Pressure	Atmospheric	Atmospheric	Atmospheric
Developed Head, psi	0.5	125	7
Drive Power, hp	40	10	40

Compressors (Reactor Vessel Argon)

Type	<u>Circulation</u>	<u>Leakage Make-Up</u>
Number	Centrifugal	Centrifugal
Argon Flow Rate, lb/h	2	1
Inlet Temperature, °F	2000	330
Suction Pressure	150	150
Developed Head, psi	Atmospheric	Atmospheric
	0.5	15

Compressors (Primary Tank and IHX)

Type	<u>Primary Tank Gas Supply</u>	<u>IHX Gas Supply</u>	<u>IHX Flow Shut Off</u>
Number	Centrifugal	Centrifugal	Centrifugal
Argon Flow Rate, lb/h	2	2	2
Inlet Temperature, °F	3400	1600	530
Suction Pressure	150	150	150
Developed Head, psi	Atmospheric	0.5	3
Drive Power, hp	0.5	3	1.5
	2.5	8	1.5

3.8.3 Principles of Operation

Prior to receiving the initial sodium charge, all of the components of the inert gas system, all of the equipment vaults, and the refueling cell are purged with nitrogen from the nitrogen storage tanks. The excess gas is vented to the atmosphere. The components of the argon subsystem are then purged with argon.

After the purity of the gases is shown to be within the prescribed limits, the first batch of sodium can be accepted in the sodium service system (see Section 3.5).

As the operating temperature of the heat transfer systems is increased, the flow rate of gases through the subsystems is increased smoothly to the normal operating conditions. Over the range of normal plant operation, operation of the inert gas system is independent of the plant power level, except for adjustment of local cover gas pressures with changes in sodium flow rate.

During plant shutdown the gases are circulated at a minimum rate which is determined by the heat loss from the relatively cool components.

3.8.4 Safety Precautions

The following are measures which must be taken to assure the safe operation of the inert gas system:

1. The purity of the gases must be continuously monitored to assure that it is within the prescribed limits,
2. The nitrogen must be sampled periodically and tested for sodium vapor and radioactivity as an indication of sodium leaks,
3. The pressure of the gas must be continuously monitored to prevent excess pressure in the primary heat transfer system and in the refueling cell, and
4. The leak rate of each of the equipment vaults must be checked periodically.

3.8.5 Maintenance

The components of the argon subsystem, which are used to handle the cover gas of the primary heat transfer system, are contaminated with radioisotopes. The components and the area where these components are located must be purged with nitrogen before removal of one of the components is attempted. After purging, a man can enter the vault using breathing apparatus (Section 3.12).

Two parallel sets of components are provided so that maintenance of one set will not require a reactor shutdown.

3.9 RADIOACTIVE GAS HANDLING SYSTEM (R.S. Palmer, P.M. Tschamper)

3.9.1 Introduction

3.9.1.1 System Functions

The functions of the radioactive gas handling system are to collect, purify, store, and control the release of toxic and radioactive gases.

3.9.1.2 Summary Description

Contaminated gases from all sources in the plant are filtered and forced into a high pressure hold-up tank. The gases are stored in the tank until the activity has decayed to an acceptable level. The gases are then released to a tall stack where they are diluted with fresh air and then vented to the atmosphere.

3.9.1.3 Design Requirements

The system must accommodate, normally, the contaminated gas produced from the following sources:

1. Purging of the reactor vessel cover gas space twice yearly. This space must be purged of its contaminated gas before the reactor vessel shield plug can be removed.
2. Leakage into the cover gas space from the buffer region of the gas seals of the penetrations through the primary containment. Double seals with a supply of fresh argon to the intermediate space are used at each penetration.

These two sources correspond to an equivalent contaminated gas flow rate of 50 cubic feet per day (at STP).

In addition, the radioactive gas handling system must accommodate the gas produced by purging the refueling cell. Purging of the refueling cell will be done rarely. It is necessary only after an accidental release of contamination in the refueling cell.

The limiting radioactive isotopes in the cover gas are expected to be Xe-133 and Kr-85. Their fission yields (lower isomers) are 6.5% and 0.3% respectively. The fuel pins are designed to vent fission gases directly to the coolant with an equivalent gas hold-up time of nine days.

The radioactive gas handling system must store the contaminated cover gas and release it to the atmosphere at a controlled rate. The maximum gas release rate has been conservatively limited to 0.1 curie per second for all radioactive isotopes released to the atmosphere. It is expected that Xe-133 and Kr-85 will account for essentially all of the released radioisotopes.

3.9.2 Detailed Description

Table 3.28 is a description of the radioactive gas handling system components.

3.9.3 Principles of Operation

Figure 3.23 is a diagram of the radioactive gas handling system. Gas from the contaminated low pressure lines of the argon system and of the nitrogen system are vented to the inlet header of the radioactive gas handling system. In addition a line with a blower is provided directly from the cover gas space of the primary sodium tank to the inlet header. This line provides rapid control of the cover gas pressure in the primary heat transfer system.

Contaminated argon from the inlet header is cooled as it passes through a nitrogen-cooled heat exchanger. The cooled argon then flows through one of two parallel filters which removes any entrained particles. The filtered argon flows into one of two parallel compressors where its pressure is increased to 300 psig. The pressurized argon flows to one of four storage tanks.

One of the storage tanks is reserved for storing the contaminated argon which results from purging the reactor vessel cover gas space. The volume of this tank is adequate to allow such purging twice a year. Contaminated gas is held in the tank for two months, and then released at a controlled slow rate.

The other three storage tanks are used, sequentially, for storing the contaminated argon which results from the continuous in-leakage past the cover gas seals. Operation of the three tanks is such that, at any given time, one tank is being filled, one tank is holding gas to allow radioactive decay, and one tank is being discharged slowly. The storage time interval for the contaminated gas is selected so that the activity concentration due to the xenon isotopes is less than that due to the krypton isotopes. The half-lives of the lower isomers of Xe-133 and of Kr-85 are 5.3 days and 10.2 years respectively. The activity of the stored gas is continually monitored. The release rate from the tanks is controlled so that less than 0.1 curies per second of activity exhaust from the stack.

Contaminated gas which is released from storage flows through a filter to the stack. This filter removes any remaining iodine and entrained particles. A header with a blower is provided to accept lightly contaminated nitrogen gas from the purging of nitrogen filled cells. The blower forces the nitrogen through the radioactive gas filters. The filtered gas flows to the base of the exhaust stack. A blower dilutes the contaminated gas stream with air. The mixture of contaminated gas and air flows up the stack and is released to the atmosphere. The activity concentration and the flow rate of the released gas stream are continuously monitored.

FIGURE 3.23 RADIOACTIVE GAS HANDLING SYSTEM
(SIMILAR TO GE DRAWING 106D3582)

TABLE 3.28

RADIOACTIVE GAS HANDLING SYSTEM
EQUIPMENT SPECIFICATIONS

Compressors

	<u>Primary Cover Gas</u>	<u>Storage Tank</u>	<u>Nitrogen Purge</u>	<u>Exhaust Stack</u>
Type	Centrifugal	Reciprocating	Centrifugal	Centrifugal
Number	2	4	2	2
Flow Rate, scfm	40	60	100	250
Inlet Temperature, ° F	150	150	150	70
Inlet Pressure, psig	-1	0	0	0
Developed Head, psi	1	300	0.2	0.3

Radioactive Gas Cooler

Type	Counter flow, finned tube, nitrogen cooled
Number	1
Heat Load	9000 Btu/h
Heat transfer area	20 ft ²
Mean Temperature Difference	78° F

	<u>Tube Side</u>	<u>Shell Side</u>
Fluid	Argon	Nitrogen
Flow Rate, scfm	120	140
Inlet Temperature	250	90
Outlet Temperature	150	150

Radioactive Gas Storage Tanks

Number	4
Capacity	500 ft ³
Operating Pressure	300 psig
Operating Temperature	150° F
Material	Carbon Steel

3.9.4 Safety Precautions

The following are measures which must be taken to assure the safe operation of the radioactive gas handling system:

1. The nitrogen atmosphere in the equipment cells must be continuously mon-

itored for leakage of the radioactive gases.

2. The temperature and pressure of the storage tanks must be continuously monitored to assure adequate removal of both compression heat and decay heat.

3. The local atmospheric conditions must be continuously measured during the release of contaminated gases from the stack.

3.9.5 Maintenance

The filters in the radioactive gas handling system must be replaced periodically. To replace a filter element, a man will enter the nitrogen-filled equipment cell, utilizing the breathing apparatus described in Section 3.12. He will use a hand operated hoist and hand tools to remove the filter element and place it into a transfer cask.

Servicing or replacement of other components will also be done directly.

3.10 ELECTRICAL SYSTEM (Bechtel Corp.)

3.10.1 Introduction

3.10.1.1 System Functions

The electrical system has the following functions:

1. Transmit the gross plant power output from the main generator to the main output transformer,
2. Receive and distribute the electrical power for use within the plant during start-up and normal operation,
3. Store energy for emergency use within the plant when a loss of outside power occurs.

3.10.1.2 Summary Description

The electrical power generation, distribution, and control systems are designed to provide maximum reliability for continuous operation. The generation system (see Figure 3.24) is conventional, utilizing modern

methods of control, excitation, and relaying. Separation of the generator from the grid system will have no adverse effect upon the plant auxiliary facilities, since automatic transfer to the standby transformer would provide an alternate source of power from the grid system. In case of simultaneous loss of generator and grid system, the auxiliary power facilities will automatically carry out a predetermined series of control functions to start up the emergency power equipment and restore power to critical plant loads, in order to maintain their operation during the interruption of normal power. Through a system of interconnections and bus tie breakers, system continuity is maintained by an automatic control system that will seek out an available source of power and switch it to any bus section that may have become separated from its own source of power. The auxiliary power system is normally supplied from the 13,800-volt auxiliary power transformer.

Power is distributed downward through a number of transformers and distribution buses. The size and numbers of buses into which this system has been separated is determined by the lower-echelon loads. Short-break power is normally provided in the 480 vac and 120 vac instrument buses. Backup power to these buses is provided by the emergency diesel generators.

The zero-break power system is designed for maximum reliability. To gain that inherent advantage of ac drives, the system is designed as an ac power system with a dc system as backup. Zero-break loads are normally fed from the normal auxiliary power system. Upon separation of any or all of the vital sources, the dc batteries will supply power to the dc bus and, through inverters, to the vital power bus. The batteries are sized to supply power to the required number of zero-break loads for a period of fifteen minutes and to the emergency lighting system for a period of one hour. This period is sufficient to enable the emergency power gen-

erators to pick up the load and re-establish power to the short-break and zero-break loads.

Through a system of interconnections and bus tie breakers, switching of power from one source to another is predetermined and fully automatic. All bus systems of 13,800 volts and less are operated grounded. System design limits ground-fault current to a low level.

3.10.1.3 Design Requirements

The electrical system is required to transmit the gross plant output, to distribute the auxiliary power, and to store energy for emergency power. These requirements must be met reliably and economically. It is also required that the equipment in the system be compatible with the electrical power distribution systems which are currently in use in the United States.

3.10.2 Detailed Description

Table 3.29 is a description of the equipment which is used in the electrical system.

3.10.3 Principles of Operation

3.10.3.1 Main Generating Subsystem

The electrical generation system is of the unit type; the main generator, main output transformer, and auxiliary power transformer are connected to form a single electrical unit.

The main generator is a 3600-rpm, three phase, 60-cycle, double-winding, hydrogen-water-cooled machine. Electrical energy is generated at 22,000 volts. The loads of each of the two windings are brought out independently to external bushings, and parallel to form a single output circuit. Generator grounding is by grounding transformer with resistance loaded secondary. Surge

suppression equipment will serve to protect the generator windings from overvoltage transients.

Generator field excitation power is provided by a direct-connected exciter alternator. A silicon diode rectifier assembly converts the ac alternator output to dc for field excitation. An amplidyne and voltage regulator in the alternator field circuit control the excitation.

Relaying is provided for generator differential, anti-motoring, over-current with voltage restraint, loss of excitation, negative sequence, overtemperature, generator ground, overcurrent, volts/cycle, and field overcurrent.

3.10.3.2 22,000 - Volt Subsystem

The generator output is delivered at 22,000 volts to the main output transformer through a liquid-cooled, isolated-phase main generator bus rated at 30,000 amperes. A 33/44/55-MVA, two-winding secondary, 22-kV/13.8-kV OA/FOA/FOA 65°C station auxiliary power transformer is tapped to the main generator bus. Disconnect links in the 22-kV tap provide for transformer isolation. This transformer serves as a main auxiliary power source to the 13.8-kV system.

3.10.3.3 13,800-Volt Subsystem

A 33/44/55-MVA, two-winding secondary, 345-kV/13.8-kV OA/FOA/FOA station auxiliary power transformer serves as a standby or startup power source to the 13,800-volt system. The 345-kV primary side of this standby transformer is connected to the high-voltage power grid. Lightning arresters provide transformer surge protection.

The 13.8-kV switchgear is indoor, metal clad, drawout type, composed of a

TABLE 3.29

DESCRIPTION OF ELECTRICAL EQUIPMENT

Generator Control Panel

Generator Excitation Control
 Main Circuit Breaker Control
 Synchronizing Equipment
 Turbine Hydraulic Control
 Protective Relays
 Loss of Field Relay
 Negative Phase Sequence Relay
 Ground Overvoltage Relay
 Voltage Imbalance Relay
 Generator Differential Relay
 Neutral Overcurrent Relay
 Unit Differential Relay
 Synchronous Check Relay
 Anti-motor Relay
 Overexcitation
 Field Ground Detector
 Auxiliary Transformer Ground O. C. Relay
 Auxiliary Transformer Differential Relay
 Auxiliary Transformer Sudden Pressure Relay
 Main Transformer Ground O. C. Relay
 Main Transformer Gas Detector Relay

Metering Equipment

Frequency meter
 Wattmeter
 Watthour meter
 Varmeter
 Frequency Recorder
 Net Power Output Recorder
 Ammeter
 Voltmeter

Generator-Isolated Phase Bus

22 kV, 30,000 ampere, 3 phase, forced cooling

Auxiliary Power Transformers Isolated Phase Bus

22 kV, 1500 ampere, 3 phase

TABLE 3.29 (Cont'd)

Auxiliary Power Transformer

22 kV/13.8 kV
 33/44/55 MVA
 Split secondary winding
 3 phase
 OA/FOA/FOA

Start-up and Stand-by Transformer

345 kV/13.8 kV
 33/44/55 MVA
 Split secondary winding with tertiary winding
 3 phase
 OA/FOA/FOA

Auxiliary Power Transformers

Three 3-phase units
 13.8 kV/480 V
 2000 kVA
 OA/FA

Auxiliary Power Transformers

Two 3-phase units
 480 V/120 - 208 V
 200 kVA
 OA

Switchgear

Two buses, each composed of the following:

13.8 kV
 250 MVA-SIC
 2000 ampere bus
 Protective relays, metering and control
 Two 2000-ampere breakers
 Eight 2000-ampere breakers

TABLE 3.29 (Cont'd)

Switchgear — Short-Break Power

Two buses, each composed of the following:

- 480 volt
- 75,000 ampere SIC
- 3000 ampere bus
- Relaying, metering and control
- Two 3000-ampere breakers
- Two 2000-ampere breakers
- Seventeen 600-ampere breakers

Switchgear — Zero-Break Power (120 VAC)

Two buses, each composed of the following:

- 120 Volt ac
- 20,000 ampere SIC
- 600 ampere bus
- 3 electrically operated 600-ampere incoming breakers
- Required feeder breakers
- 9 electrically operated 600-ampere breakers

Switchgear — Zero-Break Power (125 VDC)

Two buses, each composed of the following:

- 125 Volt dc
- 20,000 ampere SIC
- 600-ampere bus
- 9 electrically operated 600-ampere breakers
- Required feeder breakers

Switchgear—Zero-Break Power (250 VDC)

One bus, composed of the following:

- 250 Volt dc
- 20,000 ampere SIC
- 600-ampere bus
- 2 electrically operated 600-ampere breakers
- Required feeder breakers

TABLE 3.29 (Cont'd)

3 Battery Chargers

Static type
125 Volt dc
60 kw

2 Batteries

Lead-calcium, long life
125 volt
900 ampere-hour
30 min duty

3 Motor-Generator Sets (Zero Break)

100 hp, 480 volt, ac motor
100 hp, 125 volt, dc motor
75 kVA, 120/208 volt, ac generator
3 phase, 60 cycle
Shaft mounted fly-wheel

12 Motor Control Centers

480 volt
Three phase
600-ampere bus
25,000 - ampere SIC
Motor starters

Emergency Power

Diesel Gen. Unit

Three 1500-kw continuous max.
Gen. capacity 900 rpm with cooling
water heat exchanger, 2 h at 10%
overload capability, turbo - charged

Fuel Storage Day Tank

Included in Diesel Gen. Units

Fuel Storage Tank

One 50,000-gal 22 ft dia 20 ft high
1/4 in. steel plate

Diesel Fuel

Transfer Pumps and Motors

Two 100-gpm, ea. 200 ft head

2000-ampere bus, 2000-ampere incoming breakers, and 1200-ampere outgoing feeder breakers. The short-circuit rating is 250 MVA. Relaying is included for transformer differential, bus differential, motor differential, phase and ground overcurrent, motor temperature, and synchronism check.

3.10.3.4 480-Volt Subsystem (Short Break)

The 480-volt power system is composed of two separate buses rated at 3000 amperes, each fed from an auxiliary power transformer rated 2000 KVA, 13,800 to 480 volts OA/FA. Each bus is composed of two 3000-ampere, two 2000-ampere, incoming power breakers, and 600-ampere feeder breakers. Bus differential, reverse power flow, phase, and ground overcurrent relaying is included. The short-circuit rating is 75,000 amperes. Twelve motor-control centers containing the required number of motor starters, size one, two, three, and four, are provided as controllers for the various smaller motors and sodium piping heating loads, throughout the plant.

3.10.3.5 125-Volt And 250-Volt DC Subsystem (Zero Break)

Two 125-volt, 60-kW solid-state battery chargers and two 60-cell, 900 amp-h, 125-volt, lead-cadmium long-life batteries serve as power sources, through a combination of buses, to supply the total dc zero-break power requirements. Bus and feeder overcurrent protection is provided.

Included in the loads connected to the dc system are emergency lighting, fire detection system, generator and turbine essential devices, essential power-operated valves, data-logging equipment, and switchgear.

3.10.3.6 120-VAC Vital Power Subsystem (Zero Break)

The loads connected to the 120-VAC Vital Power System bus are those requiring a power source of maximum reliability, without interruption during plant disturbances. Included in the loads fed from this bus are radiation monitoring, plant communications, reactor manual control, and control-rod drive systems.

3.10.3.7 120-VAC Instrumentation Power Subsystem (Short Break)

With the exception of the vital plant instrumentation discussed under the zero-break systems, all plant instrumentation systems obtain their power from the 120-VAC instrumentation power system.

This system is composed of two separate buses, each fed from the 480-volt system through step-down transformers. A redundant transformer is provided to serve as back-up to the normal power transformers. Should either bus lose its normal power source, automatic transfer is made to the standby transformer.

3.10.3.8 Emergency Power Subsystem

In the event of loss of all auxiliary power from both the station and grid system, emergency power is provided by three 1500-kVA diesel-driven generators. All units are provided with automatic starting and switching facilities. The emergency-power diesel generators are designed to start automatically upon loss of all normal sources of power, and reach rated frequency and voltage and pick up load within 20 seconds. Manual start control is also provided. Power to start the engines is self-contained and does not depend on the availability of any other source of normal plant power. The fuel supply for each engine consists of one

main fuel storage tank and a day tank. The diesel engines use fuel which is readily available. The diesel generators are maintained in optimum condition to insure starting and operating reliability under all expected conditions.

Each generator is equipped for starting periodically to test for readiness. Each generator will start automatically on loss of voltage on the associated buses. When it reaches operating voltage and frequency, it will connect automatically to the buses. To prevent overloading of the generators, loads will be started in sequence.

The loads supplied by the diesel generators are classified in two main categories:

1. Loads required for the maximum design accident with normal power outage. Loads required immediately are started or restarted automatically upon restoration of bus voltage by the diesel generator or the normal station auxiliary power system.
2. Loads required for shutdown under non-accident conditions with power outage. These loads are started either automatically or manually.

Any two units are capable of carrying the emergency plant load. The third unit serves as a spinning reserve, to pick up the load of any unit that should fail to supply power to its assigned bus within 20 seconds. System design provides that all short-break emergency loads be energized and running 20 to 100 seconds after the initial power failure. Those critical plant loads that can tolerate no interruption are assigned to the zero-break buses and will experience no interruption of operation upon loss of normal power.

Diesel generators are connected to each of the two 480-volt buses. The third unit is placed in a standby position, capable of being automatically switched to either of the two buses. Power is distributed downward, through transformers, to each of the lower voltage buses.

The two 480-volt power buses are designated as short-break power, i. e., upon the loss of normal power and the starting of the emergency power system, power is restored to the bus within 120 to 150 seconds.

Upon demand for emergency power, all loads on the short-break buses are automatically disconnected. As the buses become re-energized from the emergency sources, selected critical elements are sequentially reconnected to the buses, in a descending order of importance.

The following buses are designated as zero-break power, i. e., upon the loss of normal power, there is no interruption of power.

1. Two 120-VAC vital power buses
2. Two 125-dc power buses
3. One 250-dc power bus.

Each zero-break vital power bus is fed by an ac-dc motor-generator set; a third motor-generator set serves as a spinning reserve, equipped with automatic switching facilities. If either vital bus should experience an outage of its own motor-generator set, the standby unit is automatically connected to the powerless bus.

Each motor-generator set is normally driven by its ac motor; if an outage should occur on the ac power source, the dc motor will, without interruption, pick up the load. The dc motor is fed from its 125-Vdc battery-

powered bus. A flywheel, rotating on the shaft of each motor-generator set, will serve to sustain the motor-generator set output during transients and switchover.

The dc zero-break system comprises two 125-volt buses and one 250-volt bus. Each 125-volt bus is supplied by a 125-volt battery floated on the bus. These batteries, arranged in series, supply the 250-volt bus and provide power for the larger dc motor loads. Batteries on the buses supply power during the interim between loss of normal power and the restoration of emergency power. Power is normally supplied to the dc buses by static battery chargers of sufficient size to supply the connected bus load and maintain the batteries at full charge.

3.10.4 Safety Precautions

All of the components of the electrical system are designed to meet the high reliability standards of the nuclear industry. Except for this fact, the electrical system is no different from that employed at modern fossil-fueled power plants.

The safety precautions which apply to the operation of this system, therefore, are those which have been already established for existing power plants. There are no special safety precautions for the operation of the electrical system which apply because the plant uses a nuclear energy source.

3.10.5 Maintenance

For most of the electrical system, the maintenance procedures are those which have been established through experience on modern fossil-fueled power plants. However, the procedures which apply to the emergency power subsystem are somewhat more stringent.

The availability of power both from the storage batteries and from the diesel driven generators must be assured whenever the plant is in operation. Consequently, no repairs or replacements may be attempted on these components until the plant is completely shutdown.

The availability of the batteries and of the diesel-driven generators must be checked frequently. The level and condition of the electrolyte in the batteries must be frequently measured and adjusted as necessary. The voltage potential of each cell of the battery must be frequently measured, and defective cells must be replaced immediately.

The dc motors which are to drive the generators of the zero-break subsystem must be checked frequently. Each dc motor is checked by cutting the power of its companion ac motor and measuring the output of the associated generator. No significant reduction of the output from the zero-break generator should occur.

Each diesel-generator set must be started periodically. The time required for the unit to start and to reach full speed is measured. Any unit which does not meet specifications must be either serviced or replaced.

The level of diesel fuel in both the day tank and the storage tank must be continuously monitored. Both tanks must be periodically checked for leaks. The fuel transfer pumps must be started periodically to assure their availability.

3.11 HEATING, VENTILATING, AND AIR CONDITIONING SYSTEM (Bechtel Corp.)

The overall plant heating, ventilating, and air conditioning system is divided into three main areas. (See Figure 3.25.) The

FIGURE 3.25 HEATING, VENTILATING, AND AIR CONDITIONING SYSTEM (SIMILAR TO BECHTEL DRAWING M-15)

first, the reactor area, contains the reactor building, the reactor auxiliaries building, and the three separate steam generator areas. The second, the turbine area, contains the turbine building, the turbine auxiliaries building, the auxiliary boiler area, and the emergency generator area. Finally, the administration area contains the administration offices, the control room, the cable spreading room, the reactor access area, and the warehouse and shop areas.

The heat source is steam coils which receive steam either from extraction lines or from the auxiliary boiler. Small self-contained electric units heat any local areas that are not furnished with heated air. A chiller unit installed inside the reactor building cools the recirculation air. Air conditioning units cool the air to the administration areas.

Three 1/2-capacity 200-ton package chillers are used to cool circulating nitrogen. These chillers are located in the reactor auxiliaries building and circulate refrigerant through the nitrogen system heat exchangers. These package chillers are sized so that during maintenance of one unit the remaining two units can handle the entire heat load. Water from the service-water system is circulated through the chillers to remove heat from the plant.

Table 3.30 is a description of the heating, ventilating and air conditioning system components.

3.11.1 Reactor Area

During normal operation, air is brought into the reactor building through one 30,000-cfm fan, where it can be passed through heating coils when required. A quick-acting remote-controlled tight-shut-off damper is located at the reactor-building air intake to isolate the reactor building when

required. Within the reactor building, four 20,000-cfm recirculation fans pass air through refrigeration coils to take up the heating load when required.

The reactor building is always maintained at a pressure slightly below ambient outside pressure; this insures that any contaminated air within the reactor building cannot leak out into the open atmosphere. During normal operation, the exhaust air is removed by two 40,000-cfm fans which mix reactor-building air with fresh air and discharge it to the atmosphere through a stack. The exhaust-air ducting is so arranged that, in the event of air contamination, the exhaust air can be routed through one of two parallel absolute filtering systems. In the event of contamination, the air supply to the reactor building is stopped and the intake sealed. One of the two 40,000-cfm exhaust fans then removes the air from the reactor building through a filter system. Air is replaced in this building through leakage past the sealing damper. With two filter trains and two exhaust fans capable of exhausting through the filters, it is assured that filtering capability will be continuously available during maintenance of equipment. If the plant is operating on emergency power, a smaller 15,000-cfm auxiliary exhaust fan is used to maintain reactor-building negative pressure and filter the exhaust air.

Ducting is arranged so that all air exhausted from the reactor building can be passed through a sodium oxide scrubber in the event of a fire in the building. The discharge from this scrubber can be passed through the filtering system if radioactive contamination exists, or otherwise passed directly to the fresh-air plenum and out the stack. By installing a sodium oxide scrubber in the system, the size of the absolute filters can be reduced substantially, since they need not remove sodium oxide. During fire conditions, all supply

TABLE 3.30

DESCRIPTION OF HEATING, VENTILATING,
AND AIR CONDITIONING SYSTEM COMPONENTS

<u>Component</u>	<u>Number</u>	<u>Description</u>
Reactor building supply fan and driver	1 1	30,000-cfm fan 15 hp-motor Remote operated quick acting seal damper
Reactor auxiliaries building and steam generator areas supply fans and drivers	2 2	40,000-cfm fans 20-hp motors
Reactor building recirculation fans and drivers	4 4	20,000-cfm fans 10-hp motors
Reactor building exhaust fans and drivers	2 2	40,000-cfm fans 25-hp motors
Reactor building auxiliary exhaust fan and driver	1 1	15,000-cfm fan 10-hp motor
Steam generator areas exhaust fans and drivers	2 2	30,000-cfm fans 15-hp motors
Turbine hall supply fans	12 12	30,000-cfm fans 15-hp motors
Turbine hall heating and cooling coils		As required complete units including circulation fans, ducting, and controls
Auxiliary boiler and emergency generator area supply fans	2 2	30,000-cfm fans 15-hp motors
Auxiliary boiler and emergency generator exhaust fan	1 1	40,000-cfm fan 20-hp motor
Administration offices supply fan	1 1	10,000-cfm fan 5-hp motor Cooling coils as required Heating coils as required
Control room supply fan	1 1	10,000-cfm fan 7-hp motor Cooling coils as required Heating coils as required

TABLE 3.30 (Cont'd)

<u>Component</u>	<u>Number</u>	<u>Description</u>
Reactor access, cool spreading, warehouse, shop supply fan	1 1	30,000-cfm fan 15-hp motor Heating coils as required
Administration offices exhaust fan	1 1	5,000-cfm fan 3-hp motor
Warehouse, shop exhaust fan	1 1	10,000-cfm fan 5-hp motor
Reactor building access exhaust fan	1 1	10,000-cfm fan 5-hp motor Remote operated quick acting seal damper
Absolute filter systems	2	Complete trains including rough filters, pre-filters, and absolute filters
Sodium oxide scrubber	1	Complete unit, 15,000-cfm air flow
Air conditioning chiller	1 1 1 1	50-ton, 62-hp motor including control panel 60-gal expansion tank Chilled water pump, 120 gpm 7-1/2-hp motor
Nitrogen chillers	3	150-ton complete package chillers including compressors, pumps and drivers tanks controls

air to the reactor building can be shut off and the building sealed. The small auxiliary 15,000-cfm exhaust fan is used to maintain a negative pressure and at the same time to permit a minimum of air flow to the fire through building leakage.

Fresh air for the reactor auxiliary building and steam generator areas is supplied by two 40,000-cfm fans. Air to one

of these fans passes through a coil which heats the air when necessary. Two 30,000-cfm fans exhaust the air from the steam generator areas to the atmosphere.

3.11.2 Turbine Area

Fresh air is supplied to the turbine building through twelve 30,000-cfm fans. This air is ducted to various areas in the

turbine building and is finally exhausted to the atmosphere through roof exhausters. Air is supplied to the turbine auxiliaries building and exhausted in the same manner. Air to the auxiliary boiler area and emergency generator area is supplied by two 30,000-cfm fans and exhausted through one 40,000-cfm fan to the atmosphere. A portion of this supply air is ducted to the turbine auxiliaries area.

3.11.3 Administration Area

Air to the administrative offices and control room is fully conditioned with heating and cooling coils, and is supplied by two 10,000-cfm fans. Part of this air is recirculated; the remainder is exhausted to the atmosphere through one 5,000-cfm fan. The air pressure in the control room is kept at a slightly higher pressure than surrounding areas to prevent dust from being blown into the control room. Supply air for the reactor-building access area, cable spreading area, warehouse, and shop is supplied by one 30,000-cfm fan. This supply air can be heated if required. Exhaust for the cable spreading area, warehouse, and shop are all discharged to the atmosphere through one 10,000-cfm fan. Exhaust for the reactor-building access area is taken by a 10,000-cfm fan which discharges directly into the reactor building. This arrangement is to insure that, in any area where possible contamination exists, the air flow is from areas of less contamination to areas of more contamination.

3.12 OTHER SYSTEMS

This section provides a summary description for each of four minor systems. A detailed description of the systems and their operation and maintenance is not provided because alternate designs for these systems would have only a small effect on the cost estimates.

3.12.1 Plant and Instrument Air System (Bechtel Corp.)

The plant compressed-air system consists of two instrument air compressors at 250 scfm and 100 psig each, and one service air compressor at 500 scfm at 100 psig. Also included in this system are after-coolers and air receivers for each compressor, a complete air dryer unit for control air, and required filters, instrumentation, and piping.

The instrument air compressors are of the non-lubricated type and discharge through aftercoolers and recovery tanks to a desiccant-type dryer which consists of two regenerative dryers operating and regenerating alternately.

The service air compressor is the lubricated type, and discharges through an aftercooler and air receiver to the plant distribution piping. Crossover piping is provided between the service air system and the instrument air system. If instrument air pressure drops below a minimum pressure, air from the service system is automatically passed through an air-oil filter and into the instrument air system.

Table 3.31 is a description of the plant and instrument air system components.

3.12.2 Communications System (Bechtel Corp.)

Four communications systems are provided in the plant:

1. An in-plant telephone system connected to the local telephone company facilities,
2. An intraplant communications system, consisting of loudspeakers and handsets located in the control room and principal work areas,

TABLE 3.31

DESCRIPTION OF PLANT AND INSTRUMENT
AIR SYSTEM COMPONENTS

<u>Component</u>	<u>Number</u>	<u>Description</u>
Compressors	2	100%-capacity, 250 scfm, 100 psig horizontal water-cooled, single-stage, double acting, non-lubricated
	2	60-hp motors, 480 volts, three-phase, 60 cycle
Receivers	3	250 ft ³ carbon steel plate, design pressure = 115 psig
Aftercoolers	2	Water cooled, 250 - scfm capacity at 100 psig
Dryer	1	Complete package, dual chamber dessicant type, 500 - scfm Capacity
Service Air Compr.	1	500-scfm, 100-psig water cooled, single-stage double acting, oil lubricated type
	1	120-hp motor, 480 V, three-phase, 60 cycle
Oil Separator	1	For 250 scfm air at 100 psig
Aftercooler	1	Water cooled unit for 500 scfm air at 100 psig

3. An evacuation alarm system providing audible coverage for the plant and external plant areas to warn of a nuclear incident or other emergency conditions, and

4. A plant fire alarm system

failure or the loss of normal ac power could cut off communication between work areas vital to plant operation or safety, or limit the coverage of the alarm systems.

3.12.3 Fire and Domestic Water Systems
(Bechtel Corp.)

The communication systems are so arranged that no single equipment or circuit

Two full-capacity fire pumps furnish the water supply for the fire mains. One

pump is driven by an electric-motor and the other is driven by a diesel engine. A jockey pump runs continuously, and, through an overflow control valve, maintains the pressure in the fire-fighting systems. When the pressure in the fire system drops, the motor-driven fire pump goes on the line automatically, and if the pressure still does not reach its nominal value (for instance, because of power failure to the motor), the diesel engine starts up automatically.

The fire protection system includes a water spray system in the area of the turbine-generator to cover such items as the oil storage reservoir and purification system, the hydrogen seal oil units and the confines of the sump, and any dikes surrounding the area under the turbine and auxiliary equipment. Readily accessible hose lines and reels are distributed throughout the station so that all areas within the station, with the exception of areas containing sodium, are within 20 feet of a nozzle attached to not more than 75 feet of hose. Yard hydrants and related hose stations are strategically located throughout the yard area.

An underground piping loop with weather-protected hose reels provides hose coverage to the outside areas of the plant. 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.

Portable CO₂ and dry chemical extinguishers are provided throughout the turbine building, administration building, and warehouse, as required.

The domestic water system consists of a pressurized water tank (10,000 gallons), a hypochlorinator, and a domestic-water transfer pump. Water is taken from lines leaving the pressurized filters and is pumped

into the domestic-water storage tank. Chlorine is added to this water by the hypochlorinator. The system is sized to provide one-half peak flow demand, which is estimated to be about 50 gpm. Supply lines carry the water to the administration, turbine building, warehouse, and reactor building areas.

Table 3.32 is a description of the Fire Fighting water system components.

3.12.4 Breathing and Cooling Air System (C. E. Boardman)

Breathing air must be supplied to full-face masks in all of the air, nitrogen and argon atmosphere zones within the reactor building. In addition, body cooling air must be supplied for suited entry into all inert gas zones within the reactor building. Trail hoses are required in all cases to supply and to remove breathing and cooling air. A vacuum header is required to carry away breathing and cooling air exhaust from the low leakage suits and the masks being used within the inert atmosphere cells. The breathing air system is isolated from the inert gas environments. Breathing air for entry into air zones within the reactor building, when face masks only are in use, may exhaust directly to the reactor building. The use of suits requiring cooling air is not anticipated for work in the air zones of the reactor building.

The principal need for low leakage suits incorporating internal cooling air for the nitrogen zones is to protect the operator from excessive gas temperatures anticipated while working near high heat loads. The use of suits in the argon atmosphere of the refueling cell is required both to protect the operator from nuclear and pyrophoric contaminants as well as to maintain the low level of oxygen and water vapor normally required in this atmosphere.

TABLE 3.32

DESCRIPTION OF FIRE FIGHTING WATER SYSTEM COMPONENTS

<u>Component</u>	<u>Number</u>	<u>Description</u>
Pumps	2	Full capacity, 1000 gpm, 200-psig discharge, multistage, horizontal split case type
	1	150-hp motor, 480 volts, three-phase, 60 cycle
Engine	1	150-hp diesel
Jockey Pump	1	50 gpm, 200-psig discharge
	1	25-hp motor, 480 volts, three-phase, 60 cycle
Diesel Day Tank	1	550 gal, carbon steel
Chemical Fire Extinguishers		
CO ₂ Fire Extinguishers		

All electrical systems associated with the breathing and cooling air supply and exhaust shall be on the zero break emergency power bus.

Figure 3.26 is a schematic diagram of the breathing and cooling air system. Table 3.33 lists the functional requirements of the system and Table 3.34 describes the components of the system.

3.13 PLANT STARTUP AND CONTROL SYSTEM

3.13.1 Introduction

3.13.1.1 System Functions

The function of the plant startup and control system is to vary smoothly and safely

the power level, pressures, temperatures, and flow rates in the reactor system and in the main heat transfer systems over the full range of plant operating conditions without exposing the equipment to excessive direct or thermal stresses.

3.13.1.2 Summary Description

The plant start-up and control system is composed of two subsystems: 1. plant startup subsystem, and 2. plant control subsystem. The startup subsystem is used over the operating range from 0% to 20% of full power. The plant control subsystem is used over the range from 20% to 100% of full power.

The startup subsystem utilizes the components and the flow paths of the main

TABLE 3.33

BREATHING AND COOLING AIR SYSTEM REQUIREMENTS

Service	Breathing and Cooling Air Supply for all zones within the Reactor Building and for new fuel storage and processing area.
Operation	Average = 6 h/day Infrequent demand = 18 h/day
Face Mask Requirements	
Delivery Pressure	90 ± 5 psig
Delivery Temperature	90° F max.
Dew Point	+5° F at 1 atm abs max
Flow Rate/Mask	Variable; 1 to 2 scfm
No. of Masks in Service	8 maximum at one time
No. of Masks	10
Suit Cooling Air Requirements	
Delivery Pressure	90 ± 5 psig
Delivery Temperature	60° F max at Suit
Dew Point	-12° F at 1 atm abs
Flow Rate/Suit	Variable; 10 to 25 scfm
Vacuum Exhaust Pressure	14 - 18 in. hg vac
Vacuum Exhaust Flow Rate	Variable; 10 to 25 scfm
Exhaust Temperature	70 to 90° F
Exhaust Air Condition	Variable to Saturation
No. of Suits in Service	4 to 8 at one time
No. of Suits	8

TABLE 3.34

BREATHING AND COOLING AIR SYSTEM
COMPONENT SPECIFICATIONS
Breathing and Cooling Air Compressor

Operation	Average = 6 h/day 18 h/day max
Number	2

TABLE 3.34 (Cont'd)

Operating Conditions	
Fluid	Air
Suction Temperature	0 to 100° F
Suction Pressure	1 atm abs
Developed Head (Est.)	150 psig
scfm/each	150
Design Conditions	
Temperature	120° F max ambient
Location	Weather proof and tornado proof building
Material	Carbon steel
Connections	Single flanged inlet and outlet
Mountings	Baseplate
Drive	Electric motor, direct connect. 3600 rpm, 80 hp
Motor and Compressor Weight Est.	7000 lbs.
Receiver	200 scf cap. (min)
Comments	Two matching compressors in parallel are required to provide reduced capacity backup. With one compressor down for servicing, the remaining unit must provide air requirements of 4 suited operators.

Breathing and Cooling Air Exhaust Vacuum Pump

Service	Breathing and cooling air exhaust for all zones within Reactor Containment Bldg
Operation	Average = 6 h/day Infrequent Demand = 18 h/day
Number	2
Operating Conditions	
Fluid	Air
Suction Temperature	70 to 150° F

TABLE 3.34 (Cont'd)

Suction Pressure scfm	20 in. hg. vac. at pump (min) Variable 4 to 150 ea.
Design Conditions	
Temperature	120° F max: ambient
Location	Inside Reactor Containment Bldg air zone
Material	Carbon steel
Connections	Single flanged inlet and outlet
Mounting	Baseplate
Drive	Electric motor, direct connections 3600 rpm, 15 hp
Cooling	Air only
Motor and Vacuum Pump Weight	1000 lbs
Comments	Two matching pumps are required to provide reduced capacity backup. With one pump down for servicing the re- maining unit must provide air require- ments for 4 suited operators.

Breathing and Cooling Air Refrigeration System

Service	Refrigeration of breathing and cooling air service in inert gas zones of reactor containment bldg.
Operation	Average = 6 h/day Infrequent Demand = 18 h/day
Number	4
Operating Conditions	
Fluid	Air
Supply Pressure	90±5 psig
Supply Temperature	70 to 100° F
Supply Dew Point	+5° F at 1 atm abs

TABLE 3.34 (Cont'd)

Exhaust Pressure	90 ± 5 psig
Exhaust Temperature	40° F
Exhaust Dew Point	-12° F at 1 atm abs
Flow Rate	30 to 100 scfm
Refrigerant	Freon
Heat Exchange	To air
Component Description	
Mounting	Portable on casters
Connections	Hose supply and exhaust
Weight	500 lbs
Motor	230 vac 1 hp
Length	28 in.
Width	26 in.
Height	41 in.

Breathing and Cooling Air Receiving Tank

Fluid	Air
No. Required	2
Capacity	50 ft ³
Operating Temperature	90° F
Operating Pressure	90 psig
Design Temperature	200° F
Design Pressure, max	150 psig
Nozzles	
Nitrogen Fill	2-1/2 in.
Nitrogen	2-1/2 in.
Material	Carbon steel
ASME Sec. VIII	ASME Sec. VIII

TABLE 3-34 (Cont'd)

Estimated Weight

800 lb

Shape

3-ft diameter × 7 ft high

Breathing and Cooling Air System Connection Points

<u>Location</u>	<u>Number</u>	<u>No. of Supply and Exhaust Headers</u>	<u>No. of Connections per Header</u>	<u>Total No. of Connections</u>
N ₂ Cell Air Locks	5	4	4	80
N ₂ Cells	5	4	4	80
Irradiated Fuel Handling Cell Air Lock	1	2	4	8
Irradiated Fuel Handling Cell	1	2	4	8
Argon Cell Air Lock	1	4	4	16
Argon Cell	1	8	4	32
		4	8	32
Reactor Bldg. Air Zone*	1	20	4	80
Total No. of Connections				336

* Supply connections only, exhaust connections and lines are not required in the reactor building air zone.

heat transfer systems. In addition, means are provided to bypass selected components in the main steam system. An oil-fired auxiliary boiler is used to supply steam for use during startup. Pony motors are provided on the main sodium pumps to circulate both the primary and the secondary sodium at 20% of full flow. An electric-motor driven pump is provided to circulate the feedwater when insufficient steam is available to operate the main boiler feed pump drive turbines. A steam separator tank is provided to ensure

that no water is carried over to the stainless steel tubes of the superheater.

The plant control subsystem utilizes digital computers and hybrid computers to control the plant parameters. One digital computer receives the plant power demand signal and generates a set-point for each of the plant's major operating parameters. This computer also receives data on the performance of the major systems, and generates revised set-points as necessary.

FIGURE 3.26 BREATHING AND COOLING AIR SYSTEM
(SIMILAR TO GE DRAWING 106D3619)

A second digital computer receives the reactor power set-point from the first computer and generates the set points for neutron flux and for sodium flow rate through each core channel. This computer also receives data on actual neutron flux, control rod position, and channel flow rate, and generates revised set points as necessary.

Hybrid computers are used to continuously measure the key parameters in analog form, to convert the analog signal to a digital signal, to compare the measured signal to the set-point, and to generate an error signal.

3.13.1.3 Design Requirements

The 1000 MWe LMFBR plant startup and control system is conceptually designed to accommodate rapid load swings over a large range, without exposing the equipment to excessive thermal stresses. The following load change rates are given as guide lines:

<u>Range</u>	<u>Load Change Rate</u>
100% to 50%	10%/min
50% to 20%	5%/min
20% to 0%	depends on reason for such action

Special attention is given to the thermal cycling of the reactor fuel and core structure. The number of cycles which are acceptable may possibly be the limiting factor rather than the maximum load change rate.

A specific plant design feature which limits the choice of control parameters is the absence of sodium flow control valves in the primary and secondary sodium loops.

The normal plant shutdown procedure is not by scram, but rather by continuous reduction of load to a minimum fractional load,

and then by a shutdown procedure which minimizes the stresses on the reactor plant.

3.13.2 Detailed Description

3.13.2.1 Plant Startup Subsystem (Bechtel Corp.)

The purpose of the plant startup system is to control the plant power level and the coolant temperatures and pressures when the plant power level is between 0% and 20% of full power. The system is used normally to increase the plant power level, smoothly and rapidly, to 20% of full power after a reactor shutdown for refueling. In addition the system is used to transfer the decay heat produced by the reactor after a scram has occurred.

Figure 3.27 is a diagram of the plant startup system. Table 3.35 is a description of the plant startup subsystem components.

3.13.2.2 Plant Control Subsystem (P. M. Tschamper)

Table 3.36 is a description of the plant control system components.

3.13.3 Principles of Operation

3.13.3.1 Plant Startup Subsystem (Bechtel Corp.)

The following is a step-by-step description of the procedure for system clean-up and for plant startup after an extended shutdown period. (After a brief shutdown, the first four steps may not be necessary). See Figure 3.27.

1. Close evaporator feed valve (EF), open feed bypass valve (FB).
2. Start motor-driven startup pump and circulate cold condensate until desired purity is achieved (25% flow).

TABLE 3.35

PLANT STARTUP SYSTEM COMPONENTS

<u>Component</u>	<u>Number</u>	<u>Description</u>
Startup Separator Tank	1	1000 psi design pressure 5-ft diameter x 25 ft long
Startup Desuperheater	1	Complete unit
Startup Feedwater Recirculation Pump	1	Motor-driven, 2800-gpm, 1000-psi discharge horizontal split case type, multi-stage
	1	2500-hp motor 13,800 volts, three-phase 60 cycle constant speed
Startup Boiler Package	1	Design pressure = 200 psig, oper- ating pressure = 150 psig, 200,000 lb/h capacity complete with fan, instruments and controls for re- mote operation
	1	20-hp condensate pump motor
	2	80-hp boiler feed pump motors
	1	Self-supporting steel stack, Gunitite lines
	1	50,000 gal, fuel storage tank, 1/4 in carbon steel plate 22 ft x 20 ft high
	2	Deaerator package including: B. F. P. s - 400 gpm at 600 TDH
	1	Condensate pump - 400 gpm at 150 ft TDH
	1	Deaerator
	1	Blow-down tank
	1	Fuel storage transfer pump, dc, 100 gpm at 100 ft head, horizontal centrifugal, 5-hp motor
	1	100 gpm at 100-ft head, horizontal centrifugal, ac, 5-hp motor

FIGURE 3.27 PLANT START-UP SYSTEM (SIMILAR TO BECHTEL DRAWING M-14)

TABLE 3.36

PLANT INSTRUMENT AND CONTROL SYSTEM EQUIPMENT LISTPlant Demand Computer

Service	Plant control and calculational purposes
Type	Digital computer with large storage capacity and remote access terminals, time sharing.
Number	1
Memory	~80 K
Function	Determines set points for all the strategic plant control parameters for a given input plant load demand. Versatile programs are permanently stored in a part of the storage drums from which it makes table look-ups in a feedback fashion. Unused memory is available for other uses.

Secondary Set-Point Controllers

Service	Set-point matching of one specific control parameter
Type	Hybrid computer
Number	9 for temperature control (one of each of the following per loop) <ul style="list-style-type: none"> - Reheated steam temperature (after mixing with bypassed steam) - Desuperheater exit temperature - IHX secondary sodium exit temperature 4 for flow control <ul style="list-style-type: none"> - Secondary sodium pump flow (one for each of 3 pumps) - Boiler feedwater flow
Digital Memory	~1 K
Analog/Digital Converter	One feedback loop, two-way analog digital conversion
Function	Receives an analog signal from measuring station, converts it into digital form, compares value with digital set point, back transforms an error signal into analog and this signal induces corrective measure on mechanical control organ.
Miscellaneous	Temperature measurements yield electrical signals. The valve actuators are either electric or pneumatic. The sodium pumps have electrically driven liquid rheostats. The boiler feedpump drive turbines have pneumatically actuated throttle valves.

TABLE 3.36 (Cont'd)

Reactor Demand Computer

Service	Reactor control and reactor data acquisition and data processing
Type	Digital computer with large storage capacity, data storage and processing capability.
Number	1
Memory	~60 K
Functions	Receives plant load demand signal from plant demand computer and the values of the ratio of loop power demand over core temperature difference. With the permanently stored programs it integrates the individual loop set points and determines the set points for the flux demand, the total required primary pump flow and signals back the major neutron flux, core channel flow, temperature and pressure readings to a data processing unit. Some of this information is a feedback signal to the main plant computer.

Flux Control Computer

Service	Control of neutron flux in reactor
Type	Hybrid computer
Number	1
Digital Memory	~3 K
Analog/Digital Converter	Multiple feedback loop, two-way analog digital conversion
Function	From all the flux measuring tubes, signals are transmitted, amplified, converted to digital and compared to the set value for the specific core location. The computer decides which shim rods have to be moved to generate the precalculated flux profiles (calculated and transmitted from reactor demand computer) and sends that signal to the appropriate control rods. Simultaneously the flux is integrated over the core and compared with the demand signal from the reactor demand computer. The latter checks the core temperature difference and resets the flux demand if it is off.

Primary Set-Point Controllers

Service	Set-point matching of one specific control parameter
Type	Hybrid computer

TABLE 3-36 (Cont'd)

Number	1 for ΔT control 3 for flow control	
Digital Memory	~1 K	
Analog/Digital Converter	One feedback loop, two way analog digital conversions.	
Function	Same as secondary set point controllers.	

3.	Bring auxiliary boiler on line.	ature. Circulate until desired purity is achieved at evaporator exit.
4.	Start one circulating water pump.	
5.	Turn steam on air ejectors, deaerator, and gland seals.	13. Decrease flow to 10%.
6.	Establish condenser vacuum.	14. Adjust evaporator extraction valve (EE) to maintain 90 psia in evaporator. Adjust spillover valve (SP) to maintain 45 psia in separator. Activate the desuperheater.
7.	Secondary sodium is circulated in isothermal mode at 350° F. Bring the reactor to criticality.	
8.	With evaporator throttle valve and bypass closed (ET and ETB), set evaporator extraction valve (EE) wide open. Close the steam admission valve (SA). The water drain valve (WD) is level controlled by the separator. Crack spillover valve (SP).	15. When separator goes dry, crack steam dump valve (SD) and steam admission valve (SA). Open both while closing spillover valve (SP). 16. Warm up turbine (on turning gear). Switch deaerator from startup boiler to separator.
9.	Adjust feedwater temperature to 250° F by the deaerator heater after reducing flow to 5%.	17. Raise steam temperature (and pressure) at 100° F per hour (with reactor power) to 800 psia and 518° F in the separator.
10.	Slowly open evaporator feed valve (EF) and close feed bypass (FB).	18. Crack evaporator throttle bypass valve (ETB) and, while opening, close steam admission valve (SA) and evaporator extraction valve (EE). Bring temperature up to 750° F.
11.	Adjust reactor power to maintain sodium exit temperature at 350° F.	
12.	Increase feedwater flow to 25% and adjust condenser bypass valve (CB) to maintain 250° F feedwater temper-	19. Switch to evaporator throttle valve (ET).

20. Roll the turbine and bring up to speed at 180 rpm/min (no hold).
21. Synchronize. Switch to turbine control valves.
22. Increase feed flow, reactor power and pick up 15% load. (Auto-extraction for feed heating).
23. Switch to boiler feed pump. Switch deaerator to extraction steam. Start other circulating pump and boiler feed pump.
24. Increase reactor power and reach rated temperature and pressure throttle conditions at 30% load.
25. Additional loading is applied by auto-control.

3.13.3.2 Plant Control Subsystem (P. M. Tschamper)

The startup procedure requires that the steam plant be brought to the 20% load condition before the main turbine is synchronized to the electrical load. Figure 3.28 shows a model of the plant, with one representative heat exchanger and pump. The control equipment and its interaction are also shown.

Figure 3.29 shows the variation with power of the controlled plant temperatures and of the sodium flow rates. Temperature control is achieved by reheat bypass, main-steam desuperheat, and sodium flow regulation. Sodium flow rate is controlled by varying the speed of the pumps. Steam leaving the superheaters is desuperheated as required to maintain permissible high pressure turbine throttle conditions. Feed-water is diverted from the flow entering the evaporator and is injected into the desuperheater.

To control the steam temperature at the throttle of the reheat turbine, a re-heater bypass is used. Unreheated steam is mixed with reheated steam. The bypass flow rate is adjusted to produce the required throttle temperature.

Figure 3.29 shows that the temperatures of primary sodium at the core outlet and at the IHX outlet (primary sodium tank inlet) are held constant. The temperature of sodium at the core outlet is maintained constant by allowing the temperature at the superheater outlet (desuperheater inlet) to increase with decreasing load. The steam temperature leaving the superheater is reduced to values suitable for the turbine by the use of a desuperheater. By maintaining constant core coolant outlet temperature with varying load, the cyclic strain loads on the core structure and fuel cladding are reduced.

The temperature of primary sodium leaving the IHX (primary vessel inlet temperature) is maintained constant by varying the primary sodium flow rate approximately linearly with plant load. It is important to keep this temperature constant since a major portion of the primary system components (primary vessel, reactor vessel, pumps, and IHX support) are exposed to IHX outlet flow. Temperature fluctuations with load could cause severe thermal stress problems in these components.

The temperature of secondary sodium leaving the IHX is seen to increase slightly with decreasing load. This temperature is controlled by varying the secondary sodium pump flow rate in a similar manner to the primary sodium flow. The flow rate change with load is approximately linear, except at high plant loads, as seen in Figure 3.29.

The resulting temperature of sodium entering the boiler is seen also to vary only slightly with load. This temperature variation with load serves to minimize thermal

FIGURE 3.28 PLANT INSTRUMENTATION AND CONTROL SYSTEM (SIMILAR TO GE DRAWING 174F130)

FIGURE 3.29 PLANT TEMPERATURE AND SODIUM FLOW AT PART LOAD

shock to the steam generator components.

To effect the control functions discussed above, a reference concept has been selected. Numerous other concepts are possible; some will be discussed in the Task III report. The particular concept selected for this Task II effort utilizes a computerized control concept which converts load demand into local demands for flow rates, temperatures, pressures and neutron flux. This concept was selected because its set-point technique means that it can bring the plant to a fractional load condition much faster than a conventional control system.

A digital plant load demand computer with a memory of about 80 K (similar to the process control computers used in oil refineries and to recent BWR control computers) receives a load demand signal from the area dispatch controller. This input initiates a calculation according to a built-in program which assigns a load demand to the loops which are currently in commission (minimum of two). The sum of those demands is compared and adjusted to the required plant demand. Once the loop demands are established, the computer proceeds to calculate the set points for the strategic control parameters (temperatures, flow rates, and core neutron flux). These set points are generally found by a "table lookup" in which the set point is given as a function of the loop load demand only. The value obtained is then transmitted to the set-point controllers, which are hybrid computers. They are capable of continuously converting a parameter measurement, which is amplified and modulated by an analog circuit, into a digital signal. This measured and converted signal is then compared to the digital set-point signal. Each of these set-point controllers are linked to a mechanical control device which influences the magnitude of the parameter which is to be adjusted. The difference between the actual parametric value and the set-point value is transformed into a signal

(analog conversion) which causes a corrective action at the mechanical control device. In order to obtain a stable response, each set-point controller must have a carefully designed electrical feedback loop which is unique to each of the valve and pump characteristics. The set point computers are small in memory and physical dimensions, since they serve only a single purpose.

The control of the reactor is different, though it still uses the set-point technique. Because of the necessary integrations of the loop conditions and of transforming that information into commands to the core, a separate computer with back-up equipment is necessary. The reactor demand computer will receive a plant demand signal from the main plant demand computer which also sets the primary pump flow controllers based on individual IHX primary sodium ΔT 's and/or loop demands. It also obtains a signal from each primary pump and from the core inlet and outlet temperature sensors. The total actual flow and the total set-point flow are calculated, as well as the actual and set-point core temperature differences. (Note that considerable reliability and redundancy has to be built into the measuring devices). With this information the computer initiates a calculation using stored programs to determine the required flux and the change in flux required to obtain the set-point conditions. For safety reasons two analog circuits are available. They are designed such that one of them calculates the anticipated flux at the demand set-point and the other one calculates the flux for the actual state. These two pieces of information go into another analog terminal which generates a signal proportional to the actual flux increment which is necessary to reach the set-point requirement.

To insure that the calculation of the actual flux is functioning properly, the output signal from the actual state flux analog

computer is transformed into a signal which can be compared to the currently measured flux. Simultaneously, the flux change which is required to reach the set point is compared to the digitally calculated flux change. With this scheme the demand increment calculation is checked. But the scheme does not rule out the initiation of an excessive control rod motion which might cause prompt criticality. Therefore, both the digital signal and the analog signal which represent the flux demand change are separately used to determine the required rod motion. After a table look-up, the signal is considered valid (in both cases) only if the required rod motion or position is safe and permissible. If the calculation or comparison shows that such a set point is unsafe, the signal is not transmitted to the flux controller. Again, only if the flux demand controller receives both signals will it consider the set-point signal as valid and will start adjusting to its value.

As the adjustment takes place, the computers will, after a short reset pause, repeat the calculation and comparison procedure, using the new current, or actual state flux conditions.

It should also be noted, that the signals to the flux demand computer are transmitted on a multi-channel basis. Two out of three coincidence is required of each signal.

The digital reactor control computer requires a memory of about 60 K. It is also used for core data acquisition and processing (with print-out capability) besides generating the flux demand signal.

Startup and shutdown of the reactor is also controlled by the main plant computer, but without such a refined set-point technique. Most of the adjustments in the 0-20% power range are exercised by remote manual adjustments from the control room.

In a scram situation the reactor is controlled separately from the normal on-line control. Since decay heat removal must continue during and after scram at a predetermined rate, a separate automatic controller with a preset circulation flow is used.

On the secondary sodium and steam side, the main plant control computer takes over the function of separating the turbine from the net following scram, isolating it and bypassing steam to the condenser, etc. Since the computer always has information available concerning the actual plant and component loads, proper conditionally seasoned control of these components during and following scram is achieved by utilizing special programs for scram shutdown.

3.13.4 Safety Precautions

The following measures must be taken to assure the safe operation of the plant start-up and control system:

1. All of the plant systems must be carefully checked before initiating a start-up, to assure that the systems are ready,
2. Key parameters must be continuously monitored during plant operation and manual override of the system must be initiated whenever anomalies are noticed, and
3. The stability of the system must be periodically checked to assure that adequate margins exist.

3.13.5 Maintenance

The support systems of the digital computers require continuous attention and adjustment. These systems are the power supply system and the air conditioning system. The maintenance procedures are those normally employed at computer installations.

SECTION IV

BUILDINGS AND STRUCTURES

4.1 CIVIL DESIGN CRITERIA (Bechtel Corporation)

The following general codes and standards govern the overall structural design:

- Uniform Building Code (UBC) 1967
- American Institute of Steel Construction (AISC) . Specification for the Design, Fabrication and Erection of Structural Steel for Buildings, Sixth Edition
- American Concrete Institute (ACI) Building Code Requirements for Reinforced Concrete (ACI 318-63)
- American Society of Civil Engineers (ASCE) Paper No. 3269, for Wind Design Requirements
- American Petroleum Institute (API) Specification No. 620 for Welded Steel Storage Tanks

4.1.1 Location

The plant site is assumed to be a 600-acre tract of land inland from the Atlantic Coast in the United States. It is located approximately four-and-one-half miles north of a small community. A large city of two million population lies approximately 60 miles northeast of the site. A major east coast shipping port is within 50 miles southeast of the site.

4.1.2 Access to the Site

The immediate area around the plant will be enclosed by a security fence, with a gate house on the access road to the south. This access road is assumed to connect with an interstate highway to the east via a county road.

The nearest railroad is a spur line of a major railroad, which ends 2-1/2 miles east of the site.

4.1.3 Description of the Environs

The plant is located on a wooded slope which forms the south bank of the river. Much of the site property consists of similarly wooded slopes rising approximately 130 feet above river elevation to level, cleared, and cultivated land. The terrain remains rough approximately 12 to 15 miles on either side of the river, and then becomes low rolling hills. There are a number of residences within one mile of the plant in the southwesterly direction, with a total of 100 people, but population within a five-mile radius is generally sparse except for two small communities to the southeast with approximately 2500 people.

4.1.4 Geology

The site is underlain by alluvial formations founded on bedrock, which is located about 50 feet below grade. The plant buildings will be founded on this soil, which is assumed to have a bearing capacity of 6000 psf.

The turbine-generator pedestal will be supported by concrete caissons carried down to bedrock.

4.1.5 Hydrology

Plant grade floor elevation is taken as 0.0 feet and will be at least four feet above the highest recorded river water elevation. Low water elevation is such that sufficient water for cooling is available at all times. A maximum cooling water temperature of 60° F has been assumed for the circulating water system design.

The normal river water elevation and water table is assumed to be at elevation

minus 8 feet at the river shoreline rising into the surrounding hills. Groundwater movement is toward the river, and there are few wells in the vicinity, so the chances of contaminating usable groundwater are extremely low. Any discharges to the river will be so controlled and diluted, both by plant cooling water and river flow, to keep activity levels well within prescribed limits.

4.1.6 Site Meteorology

The meteorological aspects of the site area are favorable for the operation of a nuclear power plant.

This is an area of very low tornado probability, but all critical structures are designed to withstand tornado loadings.

A winter outside dry-bulb design temperature of 0° F and a summer outside design temperature of 95° F dry-bulb and 65° F wet-bulb is assumed for design of the heating, ventilating, and air-conditioning systems.

For the design of the storm drainage system, a 25-year storm rainfall intensity of 2.5 inches per hour is assumed.

4.1.7 Seismology

The site is in a moderately active seismic region, but no major earthquakes have had epicenters less than 300 miles from the site. Nevertheless, the seismic design for structures important to the plant's safety features are based on the effect of the operational or design basis earthquake as defined below:

1. The operational earthquake is an earthquake having a spectrum similar to that of the El Centro, California, earthquake of May 18, 1940, except that the maximum horizontal ground acceleration shall be 0.1 g acting simultane-

ously with a maximum vertical acceleration of 0.075 g.

2. The design basis earthquake has twice the acceleration of the operational earthquake, i. e., 0.2 g horizontal acting simultaneously with 0.15 g vertical.

The applicability of the earthquakes, defined above, to individual structures, systems and equipment is covered subsequently under Section 4.1.10.

4.1.8 Wind Loading Design

All structures are designed to withstand the maximum potential loadings resulting from design winds, in accordance with standard codes and normal engineering practice.

Where structure failure could affect the operations of the plant or the continuity of reactor core cooling, structures are designed to assure safe shutdown. The effects of possible damage to these structures when subjected to pressures and missiles generated by short-term tornado wind velocities up to 300 mph will be considered.

4.1.9 Foundation Design

The geology of the area indicates that the soil and bedrock will support the loads imposed by the plant structures. No problems or restrictions beyond normal design practices are anticipated. The frost line is three feet below finished grade.

4.1.10 Plant Design Loads

The plant is designed for seismic, vertical, and lateral loads.

4.1.10.1 Seismic Loads

The AEC publication TID 7024, "Nuclear Reactors and Earthquakes" is used

as the basic design guide for seismic analyses. Definitions and criteria are as follows:

1. Class A structures, systems and components are those which are designed to ensure that, in the event of an earthquake, the plant will maintain its capability of shutting down, removing decay heat, and being safely contained, and that credible malfunctions will not tend to increase reactivity.
2. Class B structures, systems and components are those which are required to ensure normal, safe plant operation. Class B equipment includes all Class A equipment.
3. Class C structures, systems and components are those which are not required for normal, safe plant operation.
4. Class A structures, systems and components shall be designed to function following the design basis earthquake. Earthquake loads for this case shall be appropriately combined with other applicable loads, such as live loads, dead loads, wind loads, design basis accident (DBA) loads, etc., and the allowable stresses for this case shall not exceed the yield stresses of the materials, unless it can be shown by the principles of limit analysis that the combined loadings do not exceed 90% of the lower bound limit for yield collapse load.
5. Earthquake loads for Class B structures, systems and components shall be considered to arise from the operational earthquake, and shall be appropriately combined with other applicable loads, such as live load, dead loads, wind loads, DBA loads, etc., and the resultant stresses shall not exceed normal working stresses.

6. Earthquake loads and calculated stresses for Class C structures, systems and components shall be determined in accordance with the requirements of the Uniform Building Code for Zone II.

4.1.10.2 Vertical Loads

The plant is designed for live loads on floors and roofs in accordance with standard power plant design practice.

4.1.10.3 Lateral Loads

Wind loads are derived from Reference 8 as it applied to the site area and are as follows:

<u>Height (ft)</u>	<u>Pressure (lbs/sq ft)</u>	
	<u>Class A Structures</u>	<u>Class B Structures</u>
0 to 50	25	20
50 to 150	35	25
150 to 400	45	30
over 400	55	40

(One third increase in allowable stress is permitted.)

Tornado effects have been investigated on structures and equipment items related to the safe shutdown of the plant. The tornado forces applied to structures and equipment required for safe shutdown of this are as follows:

1. Rotational wind forces resulting from a tornado funnel with a velocity of 300 mph, producing a pressure of 295 lbs/sq ft.
2. Differential pressure between inside and outside of fully enclosed areas is 3 psi. Openings and blow-out panels will be considered.
3. A missile equivalent to a 4 inch x 1 foot x 12 foot long wood plank

traveling end-on at 300 mph, or a 4000-lb airborne passenger auto traveling at 50 mph at not more than 25 feet above ground, with a contact area of 30 sq. ft.

4. A torsional moment, resulting from applying the tornado load specified above on one-half of the structure and velocity equal to one-half that specified above applied to the other half of the building in an opposite direction.
5. The superstructure of the reactor building and the steam generator are designed as follows:

The metal wall siding and roof decking will be designed for the normal wind loading. When the normal wind loading is exceeded, part of the siding and decking will blow off; it is assumed that one-third stays in place and the steel framing is designed to withstand the loading caused by a 300-mph wind, acting on the full area of the concrete wall and on one-third of the original siding and decking area.

For this condition, the allowable stress in the steel frame is the yield stress. Allowable stresses in the concrete due to the tornado effect, combined with dead loads, will not exceed 90% of yield stress for the reinforcing steel and 85% of the ultimate stress of the concrete.

Missiles shall not penetrate the exterior concrete walls where damage to Class A equipment could occur.

4.2 SITE PREPARATION AND DEVELOPMENT (Bechtel Corporation)

The plant site, consisting of approximately 16 acres, is cleared and build-

ing areas cleared and grubbed. Earthwork and rough grading is assumed to be soil excavation rather than rock excavation. The excavated excess soil is hauled to a disposal area located approximately three miles from the plant site.

Storm drainage for the site consists of a system of open-surface drainage (ditches and pipe culverts), discharging into the natural drainage systems existing in the vicinity of the site. It was assumed that the building excavations required a moderate amount of dewatering.

4.2.1 Roads, Railroads, Yard Areas and Fencing

The plant is served by a two-lane access road and a single spur railroad approaching from the east (as indicated on Figure 4.1). Inside the station security fence, two-lane perimeter roads with minimum shoulders and appropriate branch roads, provide access to station buildings. Concrete walks are provided from the parking area to the administration-control building.

Station roads and parking areas are constructed of suitable base and sub-base courses, with an asphalt concrete surface course. Paved areas and roads are drained by concrete catch basins and pipe culverts. Building roof drainage will also flow to these basins and culverts.

The railroad tracks south of the reactor building serve the reactor building, the reactor auxiliaries and support building as well as the equipment laydown area in the turbine building.

A seven-foot-high security fence, topped with barbed wire and incorporating double swing gates at the road and railroad access points, is built around the plant.

4.2.2 Yard Utilities

The design of the water supply system for circulating, service and fire fighting is based on the use of river water.

Domestic water is treated in the water treatment area in the turbine building. The fire-water system consists of a 10-inch-diameter loop around the plant building, with hydrants located at approximately 250-foot intervals. Water for the fire loop will be pumped from the circulating water intake structure by means of an electric-motor-driven pump. A diesel-engine-driven pump is available for emergency standby service.

The sanitary sewer system consists of vitrified clay piping which discharges into a packaged sewage treatment plant, sized for a plant population of 80 people. After treatment, the sewage is discharged into the river near the circulating water discharge structure.

4.3 BUILDINGS, STRUCTURES, AND ARCHITECTURAL FEATURES (Bechtel Corporation)

The major buildings and structures for the LMFBR power plant are the following: reactor building; refueling cell and primary containment; reactor auxiliaries and support building; steam generator building; turbine building; control and administration building; shop and warehouse building; emergency diesel generators building; and circulating water intake and discharge structures.

The orientation and location of the buildings relative to each other are indicated on the site plan (Figure 4.1).

General arrangements of the buildings and structures are shown on Figures 4-2 through 4.8.

The equipment arrangement for the 1000 MWe power plant is the result of inter-

action between safety requirements, economic considerations, technical relationships, and structural aspects.

The relative location of the reactor building and the turbine building was determined by the minimum space requirement for developing a sound and flexible piping system between the two. The size of the turbine building is in line with current power plant design practices and accommodates all the auxiliary equipment with ample access for operation and maintenance.

Some heavy equipment requiring more frequent maintenance is located within reach of the turbine building crane. Equipment layout and piping on subsequent floors will clear the crane access hatches to these items. Sequence of the equipment is determined in such a way that piping is the simplest and shortest. This ground rule is modified somewhat where high temperatures call for flexibility in the interconnecting piping.

A few examples of the reasoning behind the location of certain equipment follows:

Water treatment equipment was placed close to the circulating water lines, this being the raw water source.

Boiler feed pumps are located as near as possible to the deaerator and feed water storage tank to shorten and simplify the critical pump suction lines.

Electrical switchgear is located above the shop and warehouse building and adjacent to the control room, within a protected area.

The closed feed water heaters are arranged over the mezzanine floor area and at the head end of the turbine on the operating deck. The location of the heaters has been selected so that pulling the tubes of any heater will not interfere with the operation of any other equipment. Tube-pulling space has also been provided for the long tubes of the condenser under the generator between ground floor and mezzanine.

FIGURE 4.1 SITE PLAN (SIMILAR TO BECHTEL DRAWING S-1)

FIGURE 4.2 PLAN AT GRADE, ELEVATION 0 feet 0 inch (SIMILAR TO BECHTEL DRAWING M-1)

FIGURE 4.4 PLAN BELOW ELEVATION 100 feet 0 inch
(SIMILAR TO BECHTEL DRAWING M-3)

SECTION E - E

PLAN AT EL. (-) 20¹-0'

PLAN AT EL. (-) 50'-0"

FIGURE 4.5 REACTOR BUILDING MISCELLANEOUS PLANS AND SECTION E-E (SIMILAR TO BECHTEL DRAWING M-4)

FIGURE 4.6 PLANT ELEVATION-SECTION A-A
(SIMILAR TO BECHTEL DRAWING M-5)

FIGURE 4.7 PLANT ELEVATION-SECTION B-B
(SIMILAR TO BECHTEL DRAWING M-6)

FIGURE 4.8 TURBINE BUILDING ELEVATION—
SECTIONS C-C AND D-D (SIMILAR TO
BECHTEL DRAWING M-7)

The operating floor has been kept free of equipment except for the main turbine generator and the high-pressure feed water heater, leaving ample lay-down space for the dismantled turbine and generator parts.

In the reactor building, the arrangement of the equipment is largely determined by environmental isolation. The primary sodium system is surrounded by nitrogen and argon systems to prevent sodium contact with outside air. Argon and nitrogen system components are arranged accordingly. Access hatches are provided to facilitate equipment removal from all levels.

The reactor auxiliary building is located adjacent to the reactor building. Sodium unloading equipment, nitrogen storage tanks, new fuel storage racks, freon cooling system, exhaust fans, and the stack are located within this building.

Administration facilities are also located adjacent to the reactor building. Locker rooms for contaminated and for uncontaminated clothing are located within this facility, with provisions for enforcing access control.

A description of structural and architectural features of each individual building or structure is given below.

4.3.1 Reactor Building (Bechtel Corporation)

The reactor building accommodates the primary sodium system consisting of the reactor, the primary tank, the primary heat exchangers, the primary pumps, and spent and new fuel storage tanks. The three secondary sodium loops are routed through the north wall of the building to the steam generator building. In addition, the building houses the primary sodium auxiliary systems including the cold traps. An equipment maintenance vault is provided at the southwest corner.

A refueling cell is provided over the reactor and the spent and new fuel storage tanks.

Covered hatches are provided in strategic locations to enable removal of equipment that is located in the vaults below the ground floor.

A 150-ton capacity electric overhead crane is provided for removing this equipment and for handling spent fuel shipping casks required for removal of radioactive components.

The building is operated at a slight negative pressure which requires all access to the building through air locks. Normal personnel access at grade is provided at the east wall through the access control area in the control-administration building. Emergency exits are provided in various locations along the west and east walls. Railroad and truck access is provided through a lock located at the south wall.

The reactor building has interior floors at elevations -20 feet, -30 feet, and -50 feet. The floor at elevation -20 feet accommodates compartments for cold traps, future hot traps, nitrogen coolers, argon cover gas coolers and waste gas compressors. The floor at elevation -50 feet provides compartments for radioactive waste gas decay tanks, primary sodium and inert gas storage tanks and other equipment.

Access to these floors is provided by stairs located in each corner of the building. For installation and access of major equipment, both during construction and during the life of the plant, floor hatches are provided in various areas. These hatches are located so as to utilize the reactor building crane for lifting equipment to the ground floor and onto railroad cars or trucks.

The dimensions of this building are 105 feet wide by 204 feet long by 105 feet high.

The superstructure consists of structural steel framing with welded steel trusses. The framing supports a metal roof deck with insulated built-up roofing and insulated metal wall siding. A concrete wall is provided to an elevation of approximately 25 feet for protection against large missiles which can be generated by tornadoes.

The site structure and building foundations are reinforced concrete. Foundations are carried down to the soil strata with the required bearing capacity. Walls and slabs below grade are waterproofed. Backfill material for these walls and slabs are granular soil suitably compacted.

4.3.2 Refueling Cell and Primary Containment (J. H. Germer)

Figure 2.16 shows the overall view of the refueling cell. It is a heavily shielded room 20 ft x 78 ft x 38.7 ft high with an additional ceiling height of 62.5 ft at one end to accommodate the raised shield plug. The cell is provided with means for remotely performing all of the normal refueling operations with direct viewing of all operations above the sodium level. The cell is filled with argon to eliminate oxidation of sodium surfaces and of equipment wetted with sodium. This argon atmosphere also permits access for refueling without contaminating the reactor cover gas.

4.3.2.1 Shield Plug

The reactor shield plug shown in Figure 2.7 is located between the reactor and the refueling cell, and is normally sealed to separate the highly radioactive reactor cover gas from the refueling cell atmosphere. Its diameter is sufficient to permit direct access to the replaceable bundles of the

reactor as well as the temporary fuel storage positions surrounding the reactor. The main body of the shield plug offers sufficient shielding to prevent excessive neutron activation of the components in the refueling cell, and when necessary, to permit a man to enter the refueling cell one day after reactor shutdown. Its construction is a continuation of the shielding of the remainder of the floor of the refueling cell. Horizontal baffles and thermal insulation are suspended from the bottom of the shield plug as over the rest of the vessel. In addition, there are projections from the bottom surface of the shield plug to a level immediately above the handles of the fuel bundles. These projections limit the possible vertical motion of the fuel bundles and guide reactor thermocouple units. These tubes, as well as extensions of the control rod drives, are guided by a grid structure which is an integral part of the shield plug. A baffle plate is located two feet below the level of the sodium, serving not only to minimize surface turbulence in the sodium, but also as an additional alignment grid for the many projections. The entire structure is made rigid by six diagonal braces.

The control rod drives are mounted on the top of the shield plug, and are installed into fixed tubes which are attached to the shield plug and which extend to the lower end of the control rod drives. The outlet thermocouples likewise are installed into fixed tubes.

The shield plug may be lifted a distance of 56.5 feet directly upward by a double-lead screw drive. Accurate guidance is required because of the many projecting components which extend below the shield plug. Guidance and lift are provided by two arms projecting from the upper surface of the shield plug. These arms extend into tracks located in slots in two opposite walls of the refueling cell. Each slot contains a large ball nut lead screw supported from bearings at the upper end of the slot, and driven in synchronism through shaft seals by an externally located drive motor.

The guidance arms are designed to provide complete restraint of the shield plug except in the vertical direction. Recirculating roller linear bearings are used. Two pairs of linear bearings are on the sides of one slot and one pair is on the sides of the other. An additional linear bearing is located at the back of each slot to provide lateral restraint. One degree of rotational restraint is provided by the synchronized lead-screw drive.

As a safety precaution, the fitting which holds the ball nut in each slot is provided with a device which would lock into the slot if the screw were to break.

Thirty clamps are provided to hold the shield plug firmly in place during operation of the reactor. These clamps are loosened and rotated out of the way by means of the overhead manipulation prior to lifting the shield plug, and are refastened after lowering the shield plug.

4.3.2.2 Intermediate Plug

The shield plug is surrounded by a semi-permanent intermediate plug which is bolted and seal-welded in place. The six fittings for the tubes from the inlet flow sensors are located in the intermediate plug as shown in Figure 2.6 and Figure 2.7. At their upper ends, these fittings penetrate the larger diameter section of the shield plug. The intermediate plug is required in order to permit installation of the reactor core structure, along with the fittings for the tubes from the inlet flow sensors. It is possible to remove the intermediate plug for reactor core structure replacement, but only with great difficulty.

4.3.2.4 Irradiated Fuel Storage Tank

The irradiated fuel storage tank has the capacity to store 174 fuel or radial blanket bundles under sodium. It is insulated and is equipped with a sodium to NaK heat

exchanger which has the capacity to remove 11.4×10^6 Btu/h (3.34 MWt) of heat at a peak operating temperature of 400° F (Section 3.7). Each fuel bundle is stored in a transfer thimble. No attempt is made to remove fuel bundles from the refueling cell until the refueling operations are all completed (Section 2.2). A storage rack in the irradiated fuel storage tank guides the fuel transfer thimbles in the same manner as the thimbles are guided in the reactor vessel. The rack is formed in five concentric circles with the individual circles of increasing diameter holding 14, 25, 35, 45 and 55 thimbles respectively.

A shield plate covers the fuel decay tank to decrease the magnitude of gamma radiation from the decay tank, and to permit effective gas sealing during fuel storage. Access through the shield plate is through five small shield plugs which form openings at the radius of each of the concentric circles. Circumferential indexing is provided rotating the shield plate.

4.3.2.5 New Fuel Storage

A tank is provided for storage of new fuel and blanket bundles in thimbles, in preparation for their transfer into the reactor vessel. This tank is similar to the fuel decay tank except that it is not normally filled with sodium. A similar storage rack holds 174 new fuel bundles.

The irradiated fuel storage tank does not have adequate capacity to store an entire core loading of 265 fuel bundles and 126 radial blanket bundles. If it ever becomes necessary to unload the entire core from the reactor, provision is made for filling the new fuel storage tank with sodium in order to store under sodium (but unshielded) all but 43 radial blanket bundles. These extra radial blanket bundles will be transferred out of the refueling cell.

4.3.2.6 Manipulators

Two overhead crane-mounted telescoping tube manipulators are provided for servicing the entire floor area of the refueling cell. The two manipulators operate from the same pair of wall-mounted rails. One of these manipulators is adapted primarily for transferring fuel bundles, in transfer thimbles, between the reactor vessel, the new fuel storage tank, and the irradiated fuel storage tank.

The other manipulator is available for general purpose operations within the refueling cell, such as removing small shield plugs from the irradiated fuel storage tank cover plate and servicing control rod drives.

The rails for the manipulators are required to cross over the wall slots which are used for raising the shield plug. This problem is solved by retracting the two rails horizontally to clear the shield plug supports. Manually operated cranks operate the retraction devices through seals in the wall of the refueling cell. When not in operation, these shafts are covered by a static seal plate to minimize leakage.

4.3.2.7 Viewing

The inside of the refueling cell can be viewed through either thick shield windows, periscopes, or closed-circuit television. Four windows are located for general surveillance of the refueling cell. Detailed viewing is provided by periscopes or by a remotely operated closed-circuit television. The television camera is mounted on a track at the sides of the cell.

4.3.2.8 Access Ports

Access ports are provided in the walls and in the ceiling of the refueling cell. These ports are of several sizes and are available for various types of operations, such as the

installation of a special-purpose manipulator or the transfer of equipment into or out of the refueling cell. The ports each have a sealed cover plate on the inside wall, and are shielded by one or more blocks of heavy concrete which can be removed from the outside of the cell.

Transfer of equipment is accomplished by removing the shielding from one of these ports and installing a tube sealed to the outside of the wall. The other end of the tube has a cover plate which permits evacuating the tube and back-filling with argon gas before removing the sealed inner cover plate. The equipment is then transferred either into or out of the refueling cell and the sealed cover plate is replaced.

A special access port is provided to permit men to enter the refueling cell. This access port has fittings for breathing tubes. (Section 3.12.4) The shield plug of the man access port is first translated on tracks into the refueling cell.

4.3.2.9 Fuel Shuffling Machine

The fuel shuffling machine is described in more detail in Section 2.2 (refueling). During reactor operation, this machine is stored in a position that permits it to be serviced if necessary. During the refueling operation it is temporarily stored as shown in Figure 2.16 at the other side of the reactor.

4.3.2.10 Primary Containment

The primary containment is by definition the boundary of the part of the reactor system which contains all of the primary coolant and cover gas. Provisions are made to assure that no fluid from the primary containment leaks out to contaminate the rest of the power plant. Portions of the sodium treatment system and cover gas system are excluded from the primary containment, where they can be isolated from the reactor by quick-operating valves.

The LMFBR primary containment is unique in that the sodium is contained in a separate tank within the primary containment. Thus the sodium tank does not constitute the primary containment. Primary containment is provided by a carbon steel vessel surrounding the primary sodium tank on the bottom, sides, and top. Insulation is provided on the inside of the primary containment within the argon cover gas atmosphere, and the outer surface of the primary containment is cooled by a forced circulation of cold nitrogen gas. In this manner, the primary containment is maintained essentially at room temperature, thus permitting the use of carbon steel and eliminating problems of differential thermal expansion in relation to the various fixed penetrations and the shield structure. Circulation of the nitrogen gas must be assured at all times to prevent overstress of the carbon steel containment vessel.

Extensions to the primary containment extend upward around the supports for the reactor, intermediate heat exchangers, and primary sodium pumps. These individual units are sealed to the primary containment extensions at the level of the top of the shield deck, completing the top part of the primary containment.

When the reactor is being refueled, the reactor is shutdown and the shield plug over the reactor is raised. At this time the function of the primary containment barrier is transferred to the walls of the refueling cell.

The barrier (tube walls, tube sheets, etc.) between the primary and secondary sodium in the intermediate heat exchangers also constitutes a part of the primary containment. A higher pressure in the secondary sodium assures that no leakage of primary sodium into the secondary sodium will occur.

Also included as part of the primary

containment are those surfaces in the argon cover gas cooling and recirculation system which have direct contact with the cover gas. This system is located in the lower part of the reactor building.

4.3.3 Reactor Auxiliaries Building (Bechtel Corporation)

This building is adjacent to and west of the reactor building. The north half of this building is a two-story high structure and the south half is a one-story high structure.

At the ground floor level, space is provided for ventilation exhaust fans and filters, a hot shop area, a new fuel storage and preparation, nitrogen storage, sodium tank car loading area, as well as tank car heating.

On the second floor at elevation 16 feet, the building accommodates the reactor building auxiliary supply fans and freon-cooling units.

The plant exhaust stack is located near the north end of this building. The stack foundation includes space for blowers, waste gas filters and for a blowdown tank.

The dimensions of this building are 46 feet wide $\times$ 203 feet long $\times$ 32 feet high and 46 feet $\times$ 203 feet $\times$ 16 feet high respectively.

In addition to normal personnel access which has been provided to the separate areas, truck access is provided to the areas for fuel and nitrogen storage. Railroad access for sodium tank cars is located at the south wall of the building.

Access to the reactor auxiliaries building from the reactor building is also through air locks, in order to maintain the negative pressure in the reactor building.

The superstructure consists of steel framing. The enclosure is insulated built-up roofing on metal roof deck and insulated metal wall siding. All framing and enclosure materials are fireproofed.

The substructure is reinforced concrete. Foundations are carried down to the soil strata with the required bearing capacity.

Walls and slabs below grade are water-proofed. Backfill material for walls and slabs are granular soil, suitably compacted.

4.3.4 Steam Generator Building (Bechtel Corporation)

This building is located between the reactor building and the turbine building and shares a common wall with each of them.

The building houses the steam generating equipment which includes the reheaters, superheaters, and the evaporators. Space is also provided for the secondary sodium pumps, the secondary sodium and the steam piping, the secondary sodium service system, and the secondary sodium expansion tanks. The building has been divided into three separate compartments with fire resistant enclosures for each.

Normal personnel access to each compartment is provided at the north wall, and emergency exits are located near the south wall of the compartments.

Transfer of equipment such as evaporators, superheaters and reheaters is made possible by the provision of a transfer aisle leading to the turbine auxiliary bay. The equipment to be transferred is lowered onto a transfer vehicle which is located on the ground floor.

This equipment is braced and supported while it is standing in a vertical position.

The equipment is transferred to the turbine auxiliary bay while still vertical. The position of the equipment is then changed from the vertical to a horizontal position by using mobile cranes or temporary hoisting rigs. It is either loaded onto a special truck and transported from the plant site, or repaired in the turbine auxiliary bay.

The superstructure of the steam generator building consists of structural steel framing and insulated built-up roofing with insulated wall siding. A concrete wall is provided up to elevation 25 feet around each secondary loop for protection of equipment and their supports against tornado effects.

The substructure is reinforced concrete. The ground floor is sloped to large carbon-steel lined pits to provide sodium drainage in case of a serious leak in the secondary sodium lines.

The foundations are carried down to the soil strata with the required bearing capacity. Walls and slabs below grade are waterproofed. Backfill material for walls and slab is granular soil, suitably compacted.

4.3.5 Turbine Building (Bechtel Corporation)

The turbine building houses the turbine-generator, the condenser, and auxiliary mechanical and electrical equipment. One electric overhead traveling 150-ton bridge crane is provided for initial installation and maintenance of turbine-generator components.

The building is located north of the steam-generator building, with the turbine-generator axis running east-west. On the south side, an auxiliary bay accommodates the auxiliary boiler and the steam piping. The steam piping is suspended from the roof. The deaerator and start-up separator are located on the roof of this bay. Normal station personnel access at grade is provided.

vided at the east and west ends of the turbine building.

This building has two elevated floors, namely the turbine operating floor at elevation 40 feet, and the partial mezzanine floors at elevation 18 feet and 20 feet respectively. The turbine-operating floor provides lay-down space for turbine and generator components. Several open or grating-covered hatches are located for transferring components and equipment to lower floors by using the overhead crane. A large hatch has been provided at the south-east corner for transferring turbine-generator components and other equipment to a railroad car.

The mezzanine floor accommodates auxiliary mechanical equipment.

The ground floor accommodates auxiliary mechanical and electrical equipment and water treatment equipment. Railroad and truck access in the south-east corner enables off-site transfer of components and equipment for repair.

Access to the elevated floors is provided by stairs near the ends and near the center of the building and connecting walkways.

The dimensions of this building are 110 feet wide by 300 feet long by 107 feet high. The dimensions of the auxiliary bay are 40 feet wide by 166 feet long by 107 feet high. The auxiliary boiler room is 40 feet wide by 59 feet long by 28 feet high.

The superstructure consists of steel purlins on welded steel roof trusses or steel roof beams, steel columns, crane girders, girts and floor framing. The elevated floors are reinforced concrete, and are isolated from the concrete turbine-generator pedestal to prevent transmission of vibrations. The ground floor is reinforced concrete on compacted granular soil.

The building enclosure is insulated built-up roofing on metal roof deck and insulated metal wall siding. No windows are provided in the exterior walls of the turbine building.

Foundations are reinforced concrete spread footings, carried down to soil strata with the required bearing capacity. These footings are tied together by means of reinforced concrete grade beams and the beams are extended to the frostline, which is assumed to be 3 feet below grade. The turbine-generator pedestal and the foundation mat are supported by concrete caissons carried down to bedrock. All backfill material is granular soil, suitably compacted.

4.3.6 Control and Administration Building (Bechtel Corporation)

The administration area of this building is one story high and the control room is two stories high. These areas are located adjacent to and east of the reactor building. The administration area provides space for an entrance lobby, toilet facilities, plant staff office space, a conference-lunch room with kitchen, a chemical and a hot laboratory room, a communications equipment area, locker rooms, and a health physics area. They are all located along the east wall of the reactor building and are arranged so as to provide controlled access and exit from the reactor building. The main personnel entrance to the plant is at the south-east corner of the administration area. An additional entrance has been provided along the east wall.

On the ground floor of the control area, space for cable spreading and rooms for batteries and motor-generator sets is provided. The second floor accommodates the control room surrounded by a viewing gallery, office space for the instrumentation engineer, and an instrument repair room.

Personnel access to the control areas

is through double doors at the south-east corner. A stairwell providing access to the second floor is also at this location.

The dimensions of the control and administration building are 75 feet wide by 227 feet long with the low roof at elevation 14 feet and the high roof at elevation 28 feet.

The superstructure is steel framed; the enclosure consists of an insulated built-up roof on a metal roof deck and insulated metal wall siding. The elevated floors are reinforced concrete. The ground floor is reinforced concrete on compacted granular soil. Windows, interior partitions and finishes are provided in accordance with good architectural practice.

The superstructure of the control area will be reinforced concrete to provide protection against tornadoes.

Building foundations are reinforced concrete spread footings, carried down to soil strata with the required bearing capacity. Spread footings are tied together by reinforced concrete grade beams which are extended to the frostline. All backfill material for grade slabs and beams is granular soil, suitably compacted.

4.3.7 Shop and Warehouse Building (Bechtel Corporation)

The shop and warehouse building is a two-story high structure. The ground floor accommodates a machine shop and a warehouse. On the second floor, at elevation 14 feet, space is provided for electrical equipment.

The building is located east and adjacent to the steam generator building, and north of the control-administration building.

The shop and warehouse areas are accessible by trucks at the east wall.

Personnel access is provided in the north-west corner. Access to the second floor is provided by a staircase at the north wall.

The dimensions of the building are 70 feet wide by 75 feet long by 28 feet high.

The superstructure consists of steel framing with insulated built-up metal roof deck and insulated metal wall siding. The east wall is reinforced concrete to provide protection against tornadoes. The second floor is reinforced concrete. The ground floor is a reinforced concrete slab on compacted granular soil.

No windows are used in the exterior walls.

Foundations are reinforced concrete spread footings, carried down to soil strata with the required bearing capacity; these footings are tied together by reinforced concrete grade beams, which extend to the frostline.

All backfill material is granular soil, suitably compacted.

4.3.8 Emergency Diesel Generators Building (Bechtel Corporation)

The emergency diesel generators building accommodates the diesel generators with their integral fuel oil day tanks. It is located adjacent to and south of the turbine building.

The building is a one-story high structure and is divided into three compartments, one for each diesel generator. The dimensions are 50 feet wide by 62 feet long by 14 feet high. Normal access to each compartment is provided at the west end of the building and an emergency exit is near the east wall.

This building is a reinforced concrete structure with a continuous spread footing carried down to soil strata using the required bearing capacity.

The ground floor is a reinforced concrete slab on backfill. All backfill material at the walls is granular soil, suitably compacted.

4.3.9 Circulating Water Intake and Discharge Structures (Bechtel Corporation)

The intake structure accommodates the main circulating water pumps, the traveling screens, stop logs, bar racks, the traveling screen wash pumps, the service water pumps, motor-driven and diesel-engine-driven fire pumps, and associated electrical equipment. The structure is located 200 feet north of the turbine building. The dimensions are 60 feet wide by 76 feet long by 40 feet high. The lowest part is reinforced concrete, with internal reinforced concrete partitions dividing the space into two circulating water pump compartments with one center auxiliary compartment for the other pumps. The structure is de-

signed to direct the flow of water into the pumps smoothly and at low velocities.

The river entrance to the intake structure is protected against scouring by large stones placed in layers. The deck of the structure is located four feet above the extreme high water level. A weatherproof, one-story enclosure is provided over the structure, consisting of a steel frame with built-up insulated roofing on a metal deck and metal wall siding. Hatch openings with watertight covers are provided in the roof, enabling removal of pump assemblies by a mobile yard crane.

The circulating water supply and discharge pipes are coated, welded buried steel pipes.

The discharge structure is a reinforced concrete hydraulic seal well providing maximum siphon recovery in the system. This structure is located approximately 1000 feet downstream from the intake structure. The river banks adjacent to the discharge structure are protected against scouring by stones appropriately placed.

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