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MASTER

1000 MWe

L M F B R

Liquid Metal Fast Breeder Reactor

FOLLOW-ON STUDY

ATOMICS INTERNATIONAL
TASK I REPORT

Volume 1

Prepared for ARGONNE NATIONAL LABORATORY

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United States Atomic Energy Commission Division of Technical Information

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ANL 1000-Mwe LMFBR
(LIQUID METAL FAST BREEDER REACTOR)
FOLLOW-ON STUDY
TASK I REPORT
VOLUME I

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I. SUMMARY

This report describes the conceptual design of a 1000-Mwe Liquid Metal Fast Breeder Reactor (LMFBR) plant, selected in Task I of the LMFBR Follow-On Study. This design will be used as the basis for more detailed design definition (Task II), further parametric studies and design evaluations (Task III), and definition of LMFBR R and D requirements (Task IV).

A. PROGRAM LOGIC

The 1000-Mwe LMFBR plant described herein is an extrapolation of the North-American-Rockwell-funded 500-Mwe FBR demonstration plant design now nearing the end of its conceptual phase. As such, it draws heavily upon the detailed engineering analyses and trade studies made for this 500-Mwe plant over the past two years.

For this conceptual design study it is assumed that the plant will be the first 1000-Mwe LMFBR to be constructed after the 500-Mwe FBR demonstration plant. The program logic is illustrated in Figure I-1. The company-funded

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Figure I-1. Relationship Between 500-Mwe Demonstration Plant and Follow-On 1000-Mwe Plant

500-Mwe plant design is based on current technology and anticipated progress in current programs. Conservative design margins are used for high reliability, and critical components will be subjected to proof-testing to reduce technical risk. A conservative safety approach has been taken in the number of sequential events required for the design basis accident (DBA) and in the resulting engineered safeguards and plant containment. The design is constrained by the requirements that the plant (1) be prototypic of a competitive 1000-Mwe LMFBR, (2) have a competitive incremental operating cost, and (3) be capable at maturity of achieving an operating availability of at least 90 %.

Based on preliminary performance and economic analysis, the follow-on 1000-Mwe LMFBR resulting from the preceding program will be competitive on an overall energy cost basis with light water reactor plants.

The development and test program for the 500-Mwe plant will demonstrate the technical performance parameters of the components of the 1000-Mwe LMFBR, since the fuel and control elements, intermediate heat exchangers, and steam generators are expected to be identical for both plants except for the quantities involved. Sodium pumps, valves, and the reactor vessel and structure are components which will be extrapolated in size, and therefore may require some additional proof-testing before inclusion in the 1000-Mwe plant. The similarities in basic plant characteristics between the 500- and 1000-Mwe plants are shown in Table I-1.

The 1000-Mwe LMFBR reference design resulting from the study reported in this document is not intended to be an optimum design, but is intended to form the basis for definition of technical problem areas and requirements for new research and development. However, as will be discussed in the section on engineering bases, the design is close to optimum in all key areas.

B. PLANT DESCRIPTION

The plant consists of a sodium-cooled fast mixed-oxide-fueled reactor, its associated heat-transfer systems, and a turbine-generator complex with supporting equipment, all housed in suitable buildings. A summary of the main plant design and performance characteristics is given in Table I-2.

TABLE I-1
PLANT CHARACTERISTICS

	500-Mwe Demonstration Plant	1000-Mwe Follow-On Plant
Reactor rating, Mwt	1250	2500
Steam conditions, psig/ ^o F/ ^o F	2400/900/900	
Net station heat rate, Btu/kwh	8420	8375
Initial specific fissile Pu inventory, kg/Mwe*	3.4	2.5
Breeding ratio	1.27	1.29
Compound doubling time, yr*	13	10
Fuel	Mixed oxides	
Maximum nominal linear heat rate, (kwt/ft) (% power)	15.0	15.0
Maximum nominal fuel clad inner surface temperature, ^o F	1137	1126
Core geometry, active height (ft):	4.16	
Equivalent diameter (ft)	5.75	7.75
Overall power coefficient 10 ⁵ (Δk/k) / (% power)	-4.19	-4.35
Number of coolant circuits	3	
Reactor outlet/inlet temperature (^o F)	1060/760	
Secondary-sodium system, hot/cold leg temperature (^o F)	950/700	950/690
Refueling cycle	6 months (twice/year)	
Refueling time (days/refueling)	7	10

*Includes ex-core requirements

TABLE I-2
PLANT DESIGN AND PERFORMANCE CHARACTERISTICS
(Sheet 1 of 2)

Nominal plant rating, Mwe	1000
Turbine-generator rating, Mwe	1066
Total reactor thermal power, Mwt (nominal)	2500
Steam conditions, psig/ ^o F/ ^o F	2400/900/900
Net station heat rate, Btu/kwh	8375
Breeding ratio	1.3
Compound doubling time, yr	10
Reactor:	
Fuel	mixed Pu-U oxide
Core dimensions:	
Fast region height, in.	50
Fast region diameter, in.	93.2
Axial blanket length, in. (each end)	12
Radial blanket height (inner/outer), in.	62/50
Lattice spacing, in.	5.389
Fuel element:	hexagonal
Number of pins	217
Clad OD, in.	0.250
Clad thickness, in.	0.015
Smeared density, % of theoretical	80
Peak nominal linear rod power, kw/ft	15
Maximum nominal cladding inner surface temperature, ^o F	1130
Overall power peaking factor	1.58
Core pressure drop, psi	100
Maximum sodium velocity, ft/sec	32.8
Number of fuel elements, (wt % fissile Pu):	
Inner	118 (11.6)
Outer	138 (16.0)
Number of radial blanket elements:	
Inner	60
Outer	66

TABLE I-2
PLANT DESIGN AND PERFORMANCE CHARACTERISTICS
(Sheet 2 of 2)

Number of control rods:	
Shim	9
Safety	6
Number of reflector elements:	72
Core loading at midcycle equilibrium, kg fissile Pu:	
Core	1760
Radial blanket	330
Axial blanket	100
Power split at midcycle equilibrium:	
Core	0.899
Radial blanket	0.069
Axial blanket	0.032
Doppler constant, T_{dk}/dT	-0.0096
Overall power coefficient at full power, $10^5 \Delta k/k/\%$ power	-4.35
Sodium heat-transfer system:	
Reactor outlet temperature, $^{\circ}F$	1060
Reactor inlet temperature, $^{\circ}F$	760
Number of circuits	3
Number of pumps/circuit	1 primary 1 secondary
Primary loop flow, gpm	76,000
Primary pump head, ft	355
Secondary loop IHX outlet temperature, $^{\circ}F$	950
Secondary loop IHX inlet temperature, $^{\circ}F$	690
Secondary loop flow, gpm	86,000
Secondary pump head, ft	250

The basic flow diagram is shown in Figure I-2. Heat generated in the reactor core and blanket is removed by liquid sodium pumped upward through the reactor, entering at 760°F and leaving at 1060. Heat is transferred from the radioactive primary circuit to the nonradioactive secondary-sodium circuit for the production of steam. Three independent sodium circuits are used, each with a primary and secondary loop. Steam at 2400 psig, 900/900°F is generated in single-wall shell-and-tube heat exchangers of the modular type. The turbine-generator plant is of conventional design, with a 3600/1800 rpm cross-compound quadruple-flow 43-in. LSB turbine. The net station heat rate is 8375 Btu/kwh.

The reactor building, shown in Figure I-3, is a reinforced concrete building with a steel liner to control leakage. The building houses the reactor and fuel-handling equipment and all portions of the primary sodium system including the intermediate heat exchangers. Double containment is provided by the reactor building and the equipment cells to protect against the accidental release of radioactivity.

Figure I-2. Simplified Flow Diagram

Figure I-3. Reactor Building

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The steam generators are located in a separate enclosed steel structure. This building also contains the steam generator sodium-side pressure relief equipment and secondary pump and storage tank. The turbine-generator building houses the prime mover and its associated equipment. The following paragraphs describe major portions of the plant briefly.

1. Reactor

The reactor consists of a central cylindrical array of fuel elements (i.e., the core) surrounded by two rows of blanket elements. The core fuel material is $\text{PuO}_2\text{-UO}_2$ and the blankets are depleted UO_2 . Tantalum control rods are used. Figure I-4 shows the reactor in elevation.

The upper closure of the reactor vessel is a double-rotating shield plug which acts as part of the fuel-handling system. The reactor is refueled on a six-month cycle with one-third of the core, one-sixth of the inner blanket, and one-ninth of the outer blanket being replaced at each refueling. Two refueling machines are used, one for in-vessel fuel handling and one for ex-vessel handling.

Figure I-4. Reactor, Elevation

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2. Sodium Heat-Transfer System

Three 1/3-capacity heat-transfer circuits are provided. The IHXs are elevated in such a way that (1) core decay-heat removal is assured by natural-convection circulation through any one of the three loops and (2) uncovering of the core as a result of a primary system leak is prevented. The sodium pumps are variable-speed free-surface single-stage centrifugal units. The main loops also contain loop isolation valves, a flow control butterfly valve to control thermal convection flow following a scram, and a check valve to prevent back-flow from the high-pressure plenum on loss of a pump. The intermediate heat exchangers are counterflow shell-and-tube type with the primary sodium on the tube side.

The evaporator, superheater, and reheater modules are all of similar design, varying only in the number of tubes per module and the tube size. These are straight shell-and-tube heat exchangers with the steam on the tube side. Relief piping is provided to vent the reaction products in the event of a large water-to-sodium leak.

3. Turbine-Generator Plant

The turbine-generator is a 3600/1800-rpm cross-compound quadruple-flow 43-in. last-stage blade unit. The rated capability is 1066 Mw at 0% makeup and 1.5 Hg absolute. Steam conditions are 2400 psig/900°F at the throttle with 900°F reheat.

C. ENGINEERING BASES

The conceptual design of the 1000-Mwe LMFBR has been developed using a systematic engineering approach which assures that all significant technical, economic, and safety factors are adequately considered in the basic engineering decision. The chief elements of this approach are the system, subsystem, and component trade studies which have been and are continuing to be performed as part of the AI500-Mwe Demonstration Plant program. These are listed in Table I-3. The trade studies marked with asterisks in the table were re-examined and updated to apply to the 1000-Mwe plant. They were selected because of their importance in setting the basic plant configuration and its performance characteristics. The trade studies selected included (1) Pool vs Loop, (2) Number of Coolant Circuits, (3) Fuel-Handling Methods, (4) Vented vs Non-vented Fuel, (5) Core Design Point, (6) System Design Point, and (7) Materials Selection. These are briefly discussed below.

1. Pool vs Loop

This basic decision, whether to place all primary heat-transfer components within the reactor vessel or to utilize a piped primary system with components separated from the reactor vessel and from each other, was given considerable attention during the early phases of the 500-Mwe Demonstration Plant design. The study made at that time was for a 1000-Mwe plant, and the updating had no effect on the basic conclusions. The pool concept would be expected to be lower in initial capital cost because of its relatively compact reactor building and markedly less extensive primary system layout. The operating availability of the loop concept is expected to be higher than that of the pool concept because of its greater maintainability. The loop concept carries with it greater flexibility for heat-transfer system modifications and improvements, and lower technical risk based on current technology. Both concepts have their particular safety advantages, and both are judged to be licensable.

TABLE I-3

TRADE STUDIES MADE TO ESTABLISH 500-Mwe CONCEPTUAL DESIGN

A. OVERALL SYSTEM

1. POOL VS LOOP*
2. PLANT CONTAINMENT
3. REHEAT VS NONREHEAT
4. PLANT CONTROL
5. AFTER-SCRAM TEMPERATURE CONTROL
6. NUMBER OF COOLANT CIRCUITS*
7. REACTOR OUTLET TEMPERATURE*
8. STEAM CONDITIONS*
9. SYSTEM ΔT 's*
10. SYSTEM ΔP 's
11. SYSTEM AVAILABILITY
12. ARGON VS HELIUM COVER GAS
13. PRIMARY/SECONDARY LOOP MATERIALS
14. SODIUM VS NAK
15. PLANT ARRANGEMENT
16. DOME ACCESS

B. REACTOR SUBSYSTEM

1. OPEN VS CLOSED LATTICE
2. VENTED VS NONVENTED FUEL*
3. CONTROL ROD (STORAGE) COOLING
4. BLANKET COOLING
5. COVER GAS CLEANUP
6. VESSEL SUPPORT METHOD
7. CORE GEOMETRY
8. CORE CLAMPING METHOD
9. CORE INSTRUMENTATION
10. VESSEL INSPECTION AND REPAIR
11. SODIUM SECONDARY CONTAINMENT
12. EMERGENCY CORE COOLING
13. SODIUM LEVEL CONTROL
14. TOP SHIELD COOLING
15. FAILED ELEMENT LOCATION SYSTEM
16. DBA PRESSURE RELIEF
17. TOP SHIELD SEALING
18. ALLOWABLE NUMBER OF ELEMENT FAILURES
19. CORE DIMENSIONS*
20. CORE ΔP
21. CORE ΔT *
22. BLANKET DIMENSIONS*
23. BLANKET RESIDENCE TIMES*
24. NUMBER OF ORIFICE ZONES IN CORE
25. FUEL PIN SIZE
26. NUMBER OF ENRICHMENT ZONES
27. NEUTRON-SHIELDING MATERIAL
28. CAVITY LINER
29. INLET LINE CONFIGURATION, TOP VS BOTTOM
30. TOP SHIELD FROST PREVENTION METHODS

C. HEAT-TRANSFER SUBSYSTEM

1. SODIUM CONTAINMENT
2. PUMP LOCATION - HOT VS COLD LEG
3. PUMP TYPE, EM VS MECHANICAL
4. PUMP SPEED CONTROL METHOD
5. VALVE TYPES
6. SYSTEM GAS REMOVAL
7. EMERGENCY COOLING
8. IHX TYPE
9. PIPING SIZE
10. NUMBER OF PUMPS
11. NUMBER OF IHX's

D. FUEL-HANDLING SUBSYSTEM

1. FUEL-HANDLING METHOD*
2. DOUBLE VS TRIPLE PLUG
3. CONCENTRIC VS ECCENTRIC CORE
4. ON-LINE VS OFF-LINE REMOVAL FROM STORAGE*
5. SINGLE VS DOUBLE BARREL EX-CORE MACHINE
6. REFUELING INTERVAL
7. DIRECT FUEL REMOVAL

E. FUEL ELEMENT

1. FUEL ELEMENT SPACER
2. FUEL ELEMENT HOLDDOWN
3. GAS PLENUM LOCATION
4. PELLET CONFIGURATION
5. FUEL SHIPPING METHODS
6. GAS PLENUM LENGTH
7. CLAD THICKNESS
8. FUEL ELEMENT SIZE
9. LINEAR POWER AND CENTERLINE TEMPERATURE
10. SMEAR DENSITY
11. PELLET/CLAD GAP
12. FUEL TYPE
13. CLAD MATERIAL
14. TOLERANCE CONTROL

F. CONTROL ROD

1. TOP VS BOTTOM LATCH
2. MAIN SEAL TYPE
3. SCRAM LATCH SEAL TYPE
4. ROD TYPES
5. COOLING METHOD
6. ABSORBER PINS VS CYLINDERS
7. IMMERSED VS NONIMMERSED BELLOWS
8. ALTERNATE SEALING METHODS
9. NUMBER OF RODS, SHIM-SAFETIES, SAFETIES
10. B₄C VS TANTALUM

G. STEAM GENERATOR

1. MODULAR VS SINGLE UNIT
2. RECIRCULATING VS ONCE-THROUGH
3. SEPARATE EVAPORATOR AND SUPERHEATER VS COMBINED
4. ALTERNATE MEANS OF EXPANSION COMPENSATION
HOCKEY STICK
BELLOWS IN-SHELL
SINE WAVE TUBE
5. MODULE SIZE
SUPERHEATER
EVAPORATOR
REHEATER
6. SUPERHEATER MATERIAL
INCOLOY VS
AUSTENITIC VS
FERRITIC (2-1/4 CR-1 MO AND
5 CR-1/2 MO)

For the first 1000-Mwe LMFBR the loop concept has been selected as the most likely to achieve technical and economic success.

2. Number of Coolant Circuits

Comparison of heat-transfer systems consisting of two, three, and four independent and identical coolant circuits showed that the least costly is one consisting of two heat-transfer circuits. The one with the highest expected plant availability consists of three circuits, since a two-circuit plant would have to be completely shut down to provide assurance of afterglow heat removal if one of the two were disabled. A plant with three or more circuits can continue in operation at reduced power with one circuit out of service. A four-circuit system is more costly than either a two- or three-circuit system, and has a slightly lower availability than a three-circuit type.

A three-circuit system was selected for the first 1000-Mwe LMFBR because of its expected availability advantage and the reasonable component size extrapolation (two times) from the 500-Mwe Demonstration Plant. For subsequent plants a two-circuit system should be seriously considered.

3. Fuel-Handling System

Because fuel handling is responsible for the bulk of scheduled shutdown time for the plant, much attention has been given to the selection of a near-optimum refueling scheme. Seven different concepts were evaluated, some with in-vessel spent-fuel storage, some with ex-vessel storage. These were:

- 1) Double-rotating plug with in-vessel fuel storage;
- 2) Double-rotating plug with out-of-vessel fuel storage;
- 3) Single-rotating plug with a rotating-arm fuel-handling machine and in-vessel fuel storage;
- 4) Fixed plug with three rotating-articulating-arm fuel-handling machines and in-vessel fuel storage;
- 5) Removable plug in hot cell with out-of-vessel fuel storage;
- 6) Single-rotating plug with a single nonrotating-articulating-arm fuel-handling machine and ex-vessel fuel storage; and
- 7) Removable plug in hot cell with in-vessel fuel storage.

Each concept was evaluated on the basis of plant availability, reliability, maintainability, safety, and capital cost.

A double-rotating plug with in-vessel fuel storage was selected for the 1000-Mwe LMFBR primarily because of the relatively high plant availability associated with it and its relatively low capital cost. The removable plug in hot cell with in-vessel fuel storage is an attractive alternate.

4. Vented vs Nonvented Fuel Elements

A study was made to evaluate the advantages and disadvantages of vented and nonvented fuel elements. The incentives for use of vented fuel elements are (1) a reduction in overall fuel assembly length which results in a lower cost reactor vessel and building as well as an improvement in handling time per fuel element, and (2) a reduction in cladding thickness which improves breeding ratio. Counterbalancing these incentives is the requirement for additional equipment to maintain an acceptable level of fission-gas impurities in the reactor cover gas, the current lack of data on the performance of vented fuels, and uncertainty in the achievable leakage rate of the reactor top shield.

The results of the study show an incentive of 0.036 mills/kwh if the plant were designed for and operated with vented fuel elements. Even in a plant designed to operate with either nonvented or vented fuel elements, there is still an incentive of 0.026 mills/kwh when vented elements are available. At the worst, if vented elements are never used, the penalty is only 0.004 mills/kwh.

These results led to the decision to design the 1000-Mwe LMFBR to permit use of either vented or nonvented fuel elements.

5. Core Design Point Selection

The core design point was selected for the conceptual design based on previous studies for the 500-Mwe Demonstration Plant. As part of the follow-on study effort, parametric studies were made to verify that the core parameters chosen were near-optimum. The parameters investigated, their reference values, and the results of the parametric studies are given in Table I-4.

As the table shows, the parametric studies verified that the reference conceptual design is near-optimum. For example, the fuel cycle cost difference between the 50-in. high reference core and the 38- to 44-in. "optimum" is in

TABLE I-4
CORE DESIGN POINT PARAMETER VALUES

Parameter	Reference Design Value	"Optimum" Value (From Parametric Studies)	Differential (Mills/kwh)
Core height, in.	50	38-44	0.005 to 0.02
Blanket height, in.	12	15	0.006
Blanket residence time, yr:			
Inner	3	2.6	0.0004
Outer	4.5	3.9	0.0004

the range of 0.005 to 0.02 mills/kwh, depending on the fuel burnup model selected. This is clearly within the limits of uncertainty in the analysis. It also underscores the need for a better understanding of the limitations on fuel life before a true optimum can be established.

6. System Design Point Selection

Selection of core ΔT , secondary heat-transfer loop ΔT , and steam temperature was also based on the 500-Mwe Demonstration Plant design, and verified by parametric studies. Results of these studies are summarized in Table I-5.

Again, it is evident that the reference design is near-optimum for core ΔT , secondary-sodium temperature, and steam temperatures.

TABLE I-5
SYSTEM DESIGN POINT PARAMETER VALUES

Parameter	Reference Design Value	"Optimum" Value (From Parametric Studies)	Differential (mills/kwh)
Core ΔT , °F	300	320-330	0.010
Secondary ΔT , °F	260	280-290	0.010
Steam temperature	900	950	0.010

7. Materials Selection

Major materials selection studies were required to select materials for the control-rod absorber, fuel cladding, and steam generator.

Choice of the control-rod material was tantalum, based on an extensive comparison of a number of different neutron-absorbing materials under the Company-funded program. The comparison narrowed the field to boron carbide and Ta. In the final comparison Ta was chosen based on the fact that no significant gas is released during the neutron-absorption process, thereby permitting a longer operational lifetime.

The structural materials normally considered in high-temperature sodium and feedwater/steam systems are austenitic steels, Types 316 and 304 SS, ferritic steel, 2-1/4 Cr-1 Mo, and low-carbon steels. Based on the experience of AI and others, and data available on operation in a sodium environment, Type 304 SS is used for the sodium piping, IHX's, pumps, auxiliary equipment, and structural members. It is also the reference choice for fuel cladding; however, Type 316 SS is being considered. The choice of 304 over 316 is based on a strength-to-cost ratio which indicates 304 is more economical.

In the steam generators, material compatibility with chlorides on the water side and sodium on the sodium side was required. In the evaporator section, ferritic steel was the logical choice due to its resistance to chloride stress corrosion, carbon transport, and relative high strength. In the superheater and reheater sections, austenitic steel has been considered in the past. However, at a temperature of 950°F ferritic steel could be used. Therefore, consideration of austenitic steels for the superheaters and reheaters was dropped because of their susceptibility to chloride stress corrosion.

In the ferritic steels 2-1/4 Cr-1 Mo was the economic choice for the steam generators. This choice was based on considering allowable stress, cost, carbon transport, and the backup provided by the usage of 2-1/4 Cr-1 Mo in previous sodium-heated steam generators. The effect of sodium on the physical properties had been investigated in various R&D programs.

The high-temperature steam piping is 2-1/4 Cr-1 Mo based on conventional steam plant practice. Low-carbon steel is used in the feedwater system based on conventional steam plant practice. Low-carbon steel is also used for

low-temperature auxiliary-sodium equipment since it is economical and compatible with sodium at low temperatures.

D. SAFETY CONSIDERATIONS

Five guidelines which provide the key safety guidance for the design were derived from the AEC General Design Criteria. These guidelines highlight the most influential safety design requirements. They cover:

- 1) Containment of radioactivity as a result of the maximum accident;
- 2) Containment of liquid metals;
- 3) Requirements for emergency core cooling;
- 4) Control and protective system requirements; and
- 5) Requirements for safety-related core characteristics.

Based on these guidelines the containment philosophy was established and dose calculations were performed. The calculations indicated large safety factors in exclusion area and low population-zone doses. The Design Basis Accident (DBA) was defined as loss-of-coolant-flow without action to reduce power.

E. DESIGN PRESELECTION

Two major design decisions were made at the outset of the follow-on study. These were selection of (1) a mixed oxide fuel, and (2) a "square" core (height and diameter limited only by considerations of performance and economics).

1. Mixed Oxide Fuel

Mixed oxide fuel was selected for the following reasons:

- 1) A sufficient technological base for this fuel system will be available in the 1976-77 period.
- 2) Operational and production experience will become available from light water reactor uranium-oxide use.
- 3) The Fast Flex Test Facility will use mixed oxide through the 1970's.
- 4) Prototypes currently planned and under construction by other countries will use mixed oxide.

2. Square Core

A "square" or "regular" geometry core is defined as a cylindrical core with height and diameter determined by considerations of performance and economics. The alternative is a "compromised" core geometry (pancake, modular, annular) which has a higher leakage and therefore a smaller positive sodium-void worth than the square core. The square geometry was selected for this study for the following reasons:

- 1) The in-core breeding ratio is higher than that of a high-leakage core.
- 2) The Doppler effect is larger in a square core.
- 3) Under normal operating conditions the overall power coefficient for a square core is more negative than for a high-leakage core.
- 4) Entire core voiding due to coolant system rupture and drain will lead to core meltdown, possible compaction, and destruction regardless of void worth.
- 5) Even in a high-leakage core, voiding will eventually result in fuel meltdown and a reactivity ramp approximately equal to that which would occur in a square core.
- 6) A square core is more sensitive to gas bubbles, thus giving advance warning in the form of a flux level signal to the control system, thereby protecting against gross fuel damage.

II. INTRODUCTION

The LMFBR Follow-On Study Task I work is part of the continuing AEC design studies of LMFBR's being managed by Argonne National Laboratory for the AEC. The objectives of the Follow-On Study Program are to identify the research and development requirements necessary for a safe, reliable, and economically competitive LMFBR and to establish the relative values of R&D programs. This information will be used by the AEC in guiding the National LMFBR Program. The study objectives will be accomplished through the development of a reference design for a 1000-Mwe plant in sufficient detail to permit determination of the engineering and operating features needed for large-scale acceptance of LMFBR's by the utility industry. The 1000-Mwe plant is assumed to be the first large commercial plant built after a demonstration plant has been built and operated, and would start operation about 1980.

The work is divided into four major tasks. In Task I the ground rules and criteria governing the study have been defined and a reference design established. Task II covers the preparation of Conceptual System Design Descriptions for the reference design at the subsystem level. Task III covers more detailed analysis of the reference design, drawings, and parametric studies of the main performance parameters. In Task IV the new data and information needed for the reference design will be defined and cost/benefit analysis will be performed to assess the relative value of improvements where cost estimates of R&D programs can be readily obtained. The R&D requirements of the demonstration plant and the relationship of the demonstration plant to the large plant will be included in Task IV.

The conceptual design of any plant has as its starting point certain basic plant requirements and criteria. The plant has to generate a specific amount of power for a certain portion of time, using a specific fuel. While the plant meets the above requirements it must do so in competition with other plants and also conform to existing local, State and Federal regulations. Since the subject plant is a hypothetical one and the study is being performed as part of a broader program, the major plant requirements and ground rules were established by the USAEC through ANL. Following are a listing and brief discussion of the plant requirements and ground rules.

- 1) The plant shall be designed as an economically competitive central-station investor-owned safe reliable nuclear power plant, based upon technology which can be reasonably assumed to be state-of-the-art in 1980.
- 2) The plant shall produce 1000 Mwe, net.
- 3) The reactor heat-transfer medium shall be liquid sodium.
- 4) The fuel requirements shall be based on a plutonium-uranium fuel cycle.
- 5) The design may include provisions for the incorporation of technological advances which are foreseeable during the life of the plant; e. g., design provisions could include means to increase specific power as improved fuel becomes available.
- 6) The design of the plant systems and components and the plant operating requirements shall be consistent with the minimum requirements of utility companies and utility company practices.
- 7) Fuel doubling times shall be computed on a compound interest basis, including total system fuel inventory.
- 8) The site assumed for the plant is one suitable for a central-station nuclear power plant of this size and type, having (1) good soil conditions, (2) adequate water availability, (3) adequate undeveloped acreage, (4) reasonable stream and air pollution ordinances, (5) accessibility by highway, railroad and navigable water systems, and (6) other features commonly sought for the installation of such plants.
- 9) The site will be assumed to be one requiring no unusual design features because of local conditions such as seismic zoning, peripheral land use, nearby habitation, and unfavorable meteorology.
- 10) The site will be assumed to be a new one and to require all of the component services and facilities inherent in a first-unit installation.

- 11) Containment design and related design features shall be based on population distribution data stated in TID-7025, Vol. I., Section 110.
- 12) The plant shall be designed to meet the utility companies' base load system demands.
- 13) The nuclear power plant shall be designed in accordance with the "General Design Criteria for Nuclear Power Plant Construction Permits," dated November 22, 1965, and all subsequent modifications and interpretations.
- 14) The reference design will be established using the subcontractor's best judgment on the probable state-of-the-art and availability of the necessary components.
- 15) Target date for start of operation for the plant is assumed to be 1980.

These overall requirements along with more detailed AI-established requirements have been incorporated into the Overall Conceptual System Design Description (OCSDD). The OCSDD forms a working guide for the plant design and has been incorporated into the report in its entirety in Section III.

III. OVERALL CONCEPTUAL SYSTEM DESIGN DESCRIPTION

A copy of the Overall Conceptual System Design Description (OCSDD) used to provide overall system design guidance during Task I is incorporated in this section. The OCSDD lists the other CSDD's which will be prepared during the Follow-On Study.

The System Design Description (SDD) provides a systematic approach to the design program. SDD's are working technical reference documents which define requirements and describe the system. They evolve through stages from Conceptual to Final as the design progresses. During the early phases of a program they are oriented toward the general plant concept and its overall design, safety, operating, and maintenance philosophy, and only the overall plant and major systems are covered. As the design progresses, more detail is included in the SDD's, and SDD's are prepared for subsystems and components.

The OCSDD included herein (CSDD-096-10-000) presents plant requirements based on the study guidelines established by ANL and on additional guidelines established by AI. It describes the plant as of the completion of Task I and is the first formal CSDD issuance. It has been reproduced in this report exactly as it was issued internally.

KEY NOUNS
System Design Description

ORIGINAL
ISSUE DATE 4-22-68

R. J. Beeley

REVISION LETTER

PAGE 1 OF
TOTAL PAGES

TITLE	Overall System Conceptual System Design Description - 1000-Mwe LMFBR Follow-on Study
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ABSTRACT

This Conceptual System Design Description presents the requirements for, and a summary description of, the 1000-Mwe LMFBR Follow-on Study plant.

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1.0 INTRODUCTION

1.1 System Functions

The primary function of the system, hereinafter referred to as the Plant, is to operate as a safe, reliable, and economical energy producer in a typical utility system. In addition, the Plant shall provide:

- 1) Design, construction, and licensing experience on a large size (1000-Mwe) plant required for demonstration of capital costs of commercial plants in the 1000- to 1500-Mwe range
- 2) Statistically significant data from operating experience on reliability, maintainability, and operability of the plant under typical central station power plant operating conditions
- 3) Statistically significant data on fuel performance and fuel cycle cost trends required for utility acceptance of LMFBRs
- 4) Construction and operating experience leading to improved commercial LMFBRs through simplified design, standardization of components, and higher reliability and availability.

1.2 Summary Description of the System

As currently envisaged, the 1000-Mwe plant shall be a self-sustaining (first unit) power station in a large (>10,000-Mwe) electric utility system. This Plant will be the first large plant built, following the Demonstration Plant. After an initial period of operation and shakedown, the Plant is expected to become a firm energy source, with power costs competitive with contemporary light water reactors.

The Plant is conveniently divided into the following major segments:

Reactor	Auxiliary Systems
Heat Transfer	Electrical
Energy Conversion	Reactor Support System
Control and Instrumentation	Land, Structures, and Conventional Systems

The key design and performance characteristics for each of these segments, as well as the underlying considerations which lead to the specific selections, are presented in detail in Section 2.0. The following description is a brief summary of Section 2.0, providing an over-view of the plant. The principal design and performance characteristics are given in Table 1.

TABLE 1
1000-Mwe LMFBR DESIGN AND PERFORMANCE
CHARACTERISTICS

Normal power (Mwe)	1000
(Mwt)	2500
Number of coolant circuits	3
Reactor inlet temperature (°F)	760
Reactor outlet temperature (°F)	1060
Number of core elements	
Fast region fuel inner zone	118
Fast region fuel outer zone	138
Radial blanket	126
Control rods	15
Fast region fuel height (in.)	50
Axial blanket length (in., each end)	12
Radial blanket length [in. (inner/outer)]	62/50
Equilibrium fissile mass (kg)	~2200
Breeding ratio	~1.3

The nuclear steam generator portion (i. e., nuclear island) of the plant utilizes a sodium-cooled FBR, fueled with mixed oxides of plutonium and uranium. Heat from the reactor core is transported by liquid sodium through primary and intermediate coolant circuits to a steam generator complex, where steam is generated at conditions similar to those for modern fossil-fueled plants. The nominal net output of the Plant described is 1000 Mwe, with a net station heat rate of about 8400 Btu/kwh.

The reactor is housed in a steel-lined concrete cavity below the operating floor of the reactor containment building. The cavity is capped with the

loading face shield, which forms a continuation of the operating floor over the reactor and provides radiation protection for operating personnel. A carbon steel dome is provided over the loading face shield for containment. The reactor vessel is suspended in the cavity from a support ring at the top of the vessel. The reactor has a high-pressure inlet coolant plenum which supplies coolant to the fast region and the blanket region. Coolant flow is proportioned with fixed orifices in the elements. Sodium flow is vertical through the reactor to a free surface pool above the core. The fuel storage region and control rods are cooled by thermosyphon from a low-pressure region below the high-pressure plenum. Coolant is supplied from bypass lines from the main lines with axial blankets at each end. Two rows of radial blanket elements are used. Separate shim control rods and safety control rods provide reactivity control, and are driven from the reactor loading face.

Heat generated in the reactor is transported to the steam generators by three parallel heat transfer circuits. Each circuit has a primary and a secondary sodium loop. The primary loops are common to the reactor, but the secondary loops are not interconnected, and are physically and hydraulically isolated from the primary system.

The primary sodium flows through the core and then through the sodium-to-sodium intermediate heat exchanger(s), where the heat absorbed in passing through the core is transferred to secondary sodium. The primary sodium becomes radioactive in passing through the core, and therefore a secondary sodium loop is employed to transport the heat to the steam generators. The secondary sodium does not become radioactive, thus allowing normal access to the steam generating facilities. In addition, with this arrangement, any pressure effects and/or sodium-water reaction products which may result from a steam generator tube leak are not propagated to the reactor.

The steam generator is arranged in three banks, one for each heat transfer circuit, with isolation valves for each bank. By arranging the steam generator equipment in this manner, a failure in one bank of heat exchangers may be isolated to leave two loops available for continued operation or removal of decay heat from the reactor, thus providing minimum interruption of power production and protection against overheating.

Steam is delivered to the turbine plant at 2400 psig/900°F/900°F. The main turbine-generator unit is a 1000-Mwe, 3600/1800 rpm cross-compound unit, with 43-in. last stage buckets.

Plant control is designed for load following between 25 and 100% of nominal plant rating. The plant control system is similar to that used in the Hallam Nuclear Power Facility (HNPF). The turbine-generator demand is met by varying sodium flow and control rod positions while maintaining the reactor ΔT approximately constant. The reactor system has a faster response to load changes than the conventional part of the plant, and therefore is not limiting.

The reactor plant includes facilities for the handling of new and spent fuel which are similar to those being employed at the FERMI and EBR-II plants. New fuel elements are transferred into reactor fuel storage inside the reactor vessel with an ex-core fuel handling machine while the plant is shut down. A second fuel handling machine, mounted on the double rotating shield plug, is used to transfer fuel from the reactor in-vessel fuel storage to core positions while the plant is shut down. Spent fuel from the core is placed in reactor in-vessel storage by the same machine. Spent fuel is removed from in-vessel storage to an ex-vessel storage tank by the ex-core fuel handling machine. Spent fuel is prepared for shipment while the plant is operating, using the ex-core fuel handling machine to transport spent fuel from the ex-vessel storage tank to a shielded cell where it is placed in the shipping cask. The ex-vessel storage system is designed to handle direct removal of fast fuel elements in the future.

The plant design incorporates features which permit the addition of a future second unit with minimum cost and disturbance to the existing unit. The major structures consist of the turbine-generator deck, adjoining steam generator structure, reactor containment building, and supporting facilities.

The Plant requirements and performance/design characteristics shall be set forth in accordance with the System Design Description Breakdown indicated in Figure 1. Components and subsystems which are not included are presumed to be either sufficiently conventional to permit their design and fabrication through the use of existing or conservatively modified standards and specifications, or which have little influence on the public safety (licensing) and operational safety (protection of utility investment) of the plant.

ANL 1000 Mwe FOLLOW-ON STUDY CONCEPTUAL SYSTEM DESIGN DESCRIPTION TREE

7692-4901

Figure 1. ANL 1000-Mwe Follow-On Study Conceptual System Design Description Tree

1.3 System Design Requirements

1.3.1 Performance

Power Output – The plant shall have a nominal net full power rating of 1000 Mwe. The rated power (100% steam flow at 2400 psig/900°F/900°F) for the plant shall correspond to the guaranteed gross output of the turbine-generator at 1.5 in. Hg absolute turbine back pressure and 0% makeup rate.

Plant Loading – The plant shall be operated as a low incremental cost station power plant (plant shall normally be loaded at rated power for 20 to 24 hr/day throughout the year), but shall have provisions for following normal load changes between 25 and 100% of rated power at 2%/min. The plant shall be designed for one such load swing per day.

Maximum Power Ramp – The plant shall be capable of sustaining an emergency change in power demand of at least 25%/min between 50 and 100% of rated power without causing a reactor trip. The plant shall be designed to accommodate four (4) such power ramps per year.

Maximum Refueling Interval – The plant shall have the capability for refueling at intervals of not less than six calendar months.

1.3.2 Availability

The generating capacity of this system will represent a significant portion of any electric system generation capability, and therefore must have availability and reliability (forced outage) factors consistent with other large generating units within typical interconnected utility grids.

Operating availability* which is defined as the percentage of time a unit is available for service whether operated or not is commonly used to describe the availability of conventional plants. This availability definition is unsuitable for engineering studies to determine trade offs between availability and other plant costs because reduced-power operation is not taken into account. Evaluation of plant availability and unavailability on an energy basis is a more suitable method. The plant energy unavailability (1-availability) is defined by the following equation:

*Analytical Report of Equipment Availability for the Seven-Year Period 1955-61, EEI Publication No. 63-42 (May 1963)

$$\text{Unavailability} = \frac{(T_s + T_f + T_E) P_o + T_i (P_o - P_i)}{8760 P_o}$$

where

T_s = total scheduled downtime during year (hr)

T_f = total downtime due to forced (unplanned) outages (hr)

T_E = total downtime due to unplanned extension of scheduled outages (hr)

P_o = design (100% steam flow) full power plant net output (Mwe)

T_i = time of partial power operation because of equipment failure (hr)

P_i = average plant net power output capability over period T_i (i. e., period when output is limited due to equipment failure) (Mwe).

The operating availability goal is 90%. The corresponding energy availability goal for a three-circuit plant is about 87%. The plant energy availability and downtime goals at plant maturity (seven years after initial commercial operation) are:

	(%)
1) Overall Availability	87.0
2) Forced Outage Rate	5.7
3) Unscheduled Extension of Refueling Shutdown (including unscheduled maintenance)	1.5
4) Schedule Downtime for Turbine Maintenance	3.8
5) Scheduled Downtime for Refueling and Concurrent Maintenance	2.0
Total	100.0

The unscheduled (Items 2 and 3) downtime goals of 5.7 and 1.5% are tentatively apportioned to the system segments as follows:

<u>Segment</u>	(%)
Reactor	1.1
Heat Transfer	3.7
Energy Conversion	1.0
Control and Instrumentation	0.3
Auxiliaries	0.5
Electrical	0.1
Support Equipment (including refueling)	0.5
Structures and Services	
Total	7.2

The unavailability factor for a given segment is defined as the percentage of a full-power year of energy that is lost, due to shutdowns or periods of limited power output which are attributable to failures in that segment.

1.3.3 Maintainability

The plant shall be designed such that scheduled maintenance can be accomplished to the greatest extent practicable while the plant is at power. Maintenance actions requiring either a hot or cold shutdown shall consider redundancy (to rotate maintenance actions) or elimination of the requirement through increased component reliability or redesign. Parts or assemblies requiring repair or replacement shall be repaired or replaced with like materials or items. Where practical, defective parts shall be repaired within the support facility.

The general provisioning policy shall be to maintain a material supply sufficient to support the plant for three months of continuous operation, for those items which are not readily available. The identification and quantity of spare parts, repairable parts, and bulk materials shall be determined after establishment of a scheduled maintenance plan.

1.3.4 Useful Life

The design service life of the plant is 30 years. During the first 20 years of service life, the plant shall meet all performance and availability requirements specified under paragraphs 1.3.1 and 1.3.2. During the last 10 years of service life, the plant shall suffer no more than a 10% reduction in availability, and no more than a 5% reduction in steaming rate.

1.3.5 Environment

Until an actual site has been selected, the following environmental conditions shall be considered design requirements, governing the conceptual design.

1.3.5.1 Topography and Site Access - The site is located on the bank of a river, 1500 ft above sea level; it is essentially level, and is clear of any obstructions or former structures. Vehicular, rail, and barge access are available to the site, with good secondary roads and railroad service available at the site boundary. Barge access, suitable for shipment of large components, such as the top shield and reactor vessel, is available in proximity to the location of the reactor building.

(SRO) AEC SENIOR REACTOR OPERATOR LICENSE
(RO) AEC REACTOR OPERATOR LICENSE

*SOME RO LICENSES DESIRABLE BUT NOT MANDATORY

TOTAL STAFF: 78
TOTAL SRO: 13
TOTAL RO: 5*

Figure 2. Organization of Operating Staff

1.3.5.2 Temperature — Summer: 90°F day dry bulb, 78°F day wet bulb
Winter: 0°F dry bulb

1.3.5.3 Hydrology — The plant is located above flood zone. Ground water is encountered at a minimum of 60 ft and a maximum of 250 ft below grade. Adequate ground water for sanitary supply and plant makeup is available within these levels.

1.3.5.4 Geology — Soil conditions at the site permit conventional excavation (no blasting) to a depth of 100 ft without requiring dewatering or sheet pilings. Allowable soil bearing values at 5 ft below grade is at least 6000 lb/ft².

1.3.5.5 Cooling Water — Plant cooling water is fresh river water at 55°F, and supply is adequate for all plant requirements.

1.3.5.6 Site Exclusion — The site exclusion requirements shall be based on population density, as per TID-7025.

1.3.5.7 Meteorology and Climatology — Site dose calculations shall be based on Pasquill Type F condition and a wind speed of 1 m/sec. For the 30-day dose calculation, a reduction of a factor of three in the dose shall be assumed for directional wind changes, and a further factor of two reduction shall be assumed for changes in wind velocity.

The site is in a 25-lb/ft² wind-pressure-map area, as defined by the Uniform Building Code. The snow load for this site is 15 lb/ft².

1.3.5.8 Seismology — Earthquake conditions at the site conform to seismic Zone 1, as defined by the Uniform Building Code, and there are no ground faults within the site boundary.

1.3.5.9 Labor — Labor availability for plant construction is adequate, but shall require transportation of 30 to 50 miles from the site to the nearest large population centers.

1.3.6 Personnel

Operation requires a multi-disciplined staff and several levels of technological training. The organization of such an operating staff can take several forms, depending upon the normal operating structure of the particular utility. A typical organization is presented in Figure 2, to indicate the overall operating personnel requirements.

1.3.6.1 Manpower Requirements – The operating and maintenance manpower requirements for a single-unit 1000-Mwe LMFBFR are estimated at 78. These requirements are based on the principle that all of the services would be provided at the station, without relying upon centralized facilities. Several of the personnel, particularly in specialist areas which will be delineated later, may not be required for the full term of plant operation, but will be needed for the first two years of operation and during any major plant changes, such as power escalations or operation with core loadings of new design.

1.3.6.2 Superintendent's Office – The plant superintendent and his assistant have overall responsibilities for the operation of the plant and matters related to plant safety and licensing. The assistant superintendent should have a substantial background in nuclear technology and plant operations, and should hold a senior reactor operator license.

1.3.6.3 Technical Staff – The technical staff is responsible for the overall review and evaluation of the plant operation, to keep management apprised of safety, operational, and plant equipment problems. The staff should be competent to react to any plant problem, provide analysis and recommendations to management, and lead outside analytical help if required.

1.3.6.3.1 Nuclear Engineer – The nuclear engineer provides analytical support in areas of reactor physics, thermal/hydraulics, and reactor control. He maintains surveillance of reactor data to determine proper operation and to obtain the required input for fuel management analyses. He provides Operations with the reactivity status of the plant, so they can recognize deviations from normal. The nuclear engineer should be a licensed senior reactor operator.

1.3.6.3.2 Health Physicist – The health physics function deals with all of the routine survey and monitoring requirements of a nuclear plant. These include pre-operational radiation surveys of the site and surroundings, continuing assessment of solid, liquid, and gaseous radioactive plant effluents, and the internal monitoring, surveying, and record keeping for protection of plant personnel. The health physicist and technician required for this function prepares general plant radiological safety practices and procedures for operations personnel as the need arises.

1.3.6.3.3 Chemist — The chemist provides surveillance of all plant fluids to assure proper purity and treatment. He assists in plant materials' problems, radiochemical analyses, and cleanup of contaminated areas or equipment.

1.3.6.3.4 Instrument Engineer — The instrument engineer is responsible for the maintenance of plant instrumentation. The instruments are maintained, calibrated, and repaired, as required, by a staff of instrumentation technicians under his direction. The group is responsible for procurement and setup of special instrumentation, as required.

1.3.6.3.5 Plant Engineer — The plant engineer is responsible for surveillance of all plant equipment, maintenance of operational histories of plant equipment, and evaluation of plant performance and efficiency.

1.3.6.3.6 Safety Specialist — For the period required to achieve reliable full-power operation, with all systems performing to design standards, a safety specialist should be available to review the plant performance and compare it to design objectives. The purpose of this review is to assure that the plant is operating to the level of safety expected, and to investigate the safety implications of those systems or components which do not operate precisely as expected, even though overall operation may not be affected. The safety specialist will probably be required for two years.

1.3.6.3.7 Training Specialist — For the period required to achieve a full complement of licensed operating personnel, a training specialist should be available to take the lead in operator training and licensing. He will give the classroom instructions, outline the training assignments, conduct written and oral examinations, and handle licensing paperwork.

1.3.6.4 Operations Staff — The plant operations staff consists of an operations supervisor, shift supervisors, and senior control operators, all of whom should be licensed senior reactor operators. Also required, as shown in Figure 2, are control operators, equipment operators, and equipment attendants. The control operators should be licensed reactor operators, as should some of the equipment operators who do not require licenses but who are needed to fill licensed operator vacancies. This operations staff is responsible for the operation of the plant according to established procedures, including periods of reactor shutdown and refueling.

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

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

1.3.6.5 Maintenance Staff – The maintenance staff consists of a supervisor and a foreman, directing the work of mechanics, electricians, and machinists. Their responsibility is to keep the plant and its equipment in good operating condition, and to provide skills for modifications and special tests or operations.

1.3.6.6 Administrative Services Staff – This staff includes a supervisor over the office, store facilities, plant protection, and janitorial staffs.

1.3.6.7 Reactor Safety Committee – The Reactor Safety Committee, comprised of specialists in areas of plant design and operation, is charged with the independent review and evaluation of planned reactor operation and of special tests or operations. Deviations from normal operation, and incidents having safety significance, are immediately brought to the attention of this committee. They should also review and approve procedures, license changes, and emergency plans.

Those parts of the organization that should be represented on the committee are the operating staff, the technical staff, and station management.

1.3.7 Safety

1.3.7.1 Nuclear Safety – The plant shall be designed in accordance with the principles of nuclear safety and accident prevention set forth in Safety Criteria TI-096-13-015 of latest revision.

1.3.7.2 Radiation Control – The exposure levels in occupied areas during reactor operation and shutdown shall be limited to values consistent with the Code of Federal Regulations, Title 10, Part 20.

1.3.7.3 Reactor Site — The reactor site and siting shall be consistent with the requirements of Code of Federal Regulations, Title 10, Part 100.

1.3.7.4 Radioactive Effluent Control — The operations personnel and the general public shall be protected from radiation and radioactive effluents in accordance with the Code of Federal Regulations, Title 10, Parts 20 and 100. Where unfavorable environmental conditions can be expected to require limitations upon the release of operational radioactive effluents to the environment, holdup capacity shall be provided as required for retention of gaseous, liquid, or solid effluents.

1.3.7.5 Industrial Hygiene — Air concentrations of toxic materials shall not exceed permissible levels, in accordance with applicable State and Federal Regulations. Equipment necessary to provide good industrial hygiene practices shall be provided. Waste disposal facilities, including both radioactive and non-radioactive materials, to prevent environment pollution shall be provided.

1.3.7.6 Fire Protection — Fire protection shall be provided in accordance with the codes and standards of the National Fire Protection Association.

1.3.8 Design

The following broad criteria shall guide the design of the plant in the following order of priority:

- 1) The estimated energy costs must be competitive with contemporary light water reactors of the same size range.
- 2) Operating conditions, hardware configuration, and material selection shall be based on expected state-of-the-art or conservative extrapolation from the Demonstration Plant Program and current development programs, with primary emphasis on reliability and availability.
- 3) Plant design shall be based on a realistic approach, which nevertheless would assure the licensing of the plant with minimum delays in schedule.
- 4) The plant must be of a design concept (including all critical components) which can be extrapolated from the Demonstration Plant Program, with minimum additional Research and Development work and component testing.

1.3.8.1 Reactor Segment – The reactor shall be of the sodium-cooled fast spectrum type, operating on a U-Pu fuel cycle. The reactor shall be designed for operation with nonvented, stainless steel clad (Type 304 or Type 316), mixed U-Pu oxide fuel. Provisions shall be incorporated to permit future modification for operation with vented fuel. Reactor performance shall be limited by the following conditions:

- 1) Fuel at hot spot shall be below the melting point at the maximum anticipated overpower condition.
- 2) Core region target average burnup shall be 75 Mwd/kg (U + Pu).
- 3) Power coefficient of reactivity shall be negative throughout the core under all credible conditions over the power operating range.
- 4) In-core integrated average cladding temperature at the hot spot shall be limited to 1200°F at 100% power.
- 5) It is assumed that fuel will be reprocessed off-site, in a central aqueous reprocessing plant.

The reactor shall incorporate neutron and gamma ray attenuating and shielding provisions, as required to assure the integrity of all permanently installed components for 30 years. Considerations shall be given to designs which permit removal, repair and/or replacement of critical components. Such capability shall not compromise the reliability of the plant during normal operations; but, at the same time, it shall not require more than one year for such low-probability operations.

The plant shall be conservatively designed to produce 100% steam flow at rated steam conditions (~2500 Mwt).

1.3.8.2 Heat Transfer Segment – The heat transfer segment shall remove the heat generated in the heat source and, via primary and intermediate sodium loops, transfer the heat (power) to steam. The heat transfer segment shall also provide a positive and reliable means of removing decay heat from the reactor under any credible set of emergency conditions. The latter requirement shall be met in conjunction with the auxiliaries segment.

The heat transfer segment shall consist of three similar circuits, connected in parallel for core heat removal. Each circuit shall have a reactor

coolant loop (primary loop) and an intermediate coolant loop (secondary loop). Each primary coolant loop shall employ one or more main coolant pumps, one or more intermediate heat exchanger units, and the required flow control devices for normal control, low-flow control, and loop isolation. Each secondary coolant loop shall employ one or more main coolant pumps, one or more steam generator units or banks, and the required flow control devices for normal control, low-flow control, and steam generator isolation.

The temperature gradients across each loop shall be nominally constant, over the desired load-following power range. Variations in power demand shall be accommodated through variation in the main coolant circulation rate, with some trim in temperatures. Valve operators, other than the drive train, shall be accessible for maintenance without required sodium decay. Design of all components and subsystems shall be with conservative engineering factors of safety, to assure rated steam flow corresponding to the ~2500-Mwt rated power output, but at no time should be less than that required to meet the steam flow corresponding to the turbine name-plate rating at design pressure.

Capability shall be provided for positive and reliable means of removing decay heat under all anticipated abnormal and credible accident conditions, including isolation from main condensers, loss of normal power to reactor recirculation pumps, and partial loss of coolant from the system between leak discovery and isolation. All components containing radioactive sodium shall be located within nitrogen-filled and appropriately shielded cells.

The reliability and maintainability for the heat transfer system shall be consistent with the availability allocation under Paragraph 1.3.2. Steam generator modules shall be grouped into multiple banks, with primary emphasis on reliability and maintainability. Provision shall be made for isolation and sodium drainage of any one of these banks with the remainder of the plant in operation.

1.3.8.3 Energy Conversion Segment – The power generation segment shall convert the heat energy in the superheated steam from the heat transfer segment into electrical energy, provide a heat sink to reject the low-level heat from the turbine exhaust, and provide for economical feedwater regenerative heating and treatment. The turbine exhaust back pressure shall be 1.5 in. Hg nominal. The main turbine-generator set shall be conservatively rated for a nominal net electrical output of ~1,000,000 kwe with 2400 psig/900°F/900°F steam conditions.

1.3.8.4 Control and Instrumentation Segment — The plant control and protective system segment shall provide for:

- 1) Regulation of the plant power level in accordance with requirements under Section 1.3.1
- 2) Control of the plant, by automatic and/or manual means, as required, during startup and normal shutdown operation
- 3) Protect the plant from damage as a result of abnormal operating conditions.

The plant control system shall provide the functions necessary to establish the set point levels for power generation in the reactor and to control the transfer of this energy to steam, in accordance with electric power demand. As power demand changes, signals from the generator will be transmitted to the reactor for a corresponding change in the generation of fission energy. Under conditions of startup, and normal and emergency shutdown, the control system shall maintain coolant flows and temperatures within design limits, maintain steam pressure and temperatures within design limits, and control condensate levels in the condenser hot well and feedwater heaters.

The following specific requirements shall further guide the design of the plant control system:

- 1) Steam pressure shall not exceed 130% of rated at any time. Accumulated time in excess of 105% of rated power shall not exceed 12 hr/year.
- 2) Main or reheat steam temperature shall not exceed rated +25°F more than 400 hr/year; shall not exceed rated +50°F more than 15 min continuously, nor more than 80 hr/year. It shall be a design objective that no "steady-state point"* shall exceed rated +100°F under any operating or casualty condition.

*A point in time is called steady state for the heat transfer system if, during the preceding and subsequent 1/2 min, steam temperatures (at the main and reheat headers) do not vary by more than 7°F, steam pressure by more than 10 psi, and power by more than 1/4%.

- 3) The plant shall be capable of maintaining "steady-state intervals"^{*} of at least 4 hr (provided feedwater temperatures do not vary more than 10°F).
- 4) Under normal load following, steam temperatures shall not vary more than 25°F, and initial pressure not more than 100 psi.
- 5) Provisions shall be made for testing of the plant to ASME Power Test Codes.

The reactor control system shall provide the functions necessary to control the fission energy generation in the reactor and to respond to the demand signals from the plant control center (plant load computer). The neutron flux from the reactor shall be monitored from source level to 150% of full nuclear power. Steady-state and transient temperature measurements shall be monitored and maintained within design limits. The reactor control shall have the capability of controlling the reactor from a cold startup at a rate of temperature increase as required by the turbine plant ($\sim 100^\circ\text{F/hr}$).

The plant protective system shall prevent any damage which would result in uncontrolled release of radioactivity from the reactor system. In addition, it shall provide the functions necessary to protect the plant, equipment, and personnel during abnormal operations. The protective action shall be applied automatically, and shall either reduce power to a safe operating level or shut the plant down, whenever unsafe conditions are detected or exist. This system shall protect the plant against damage from thermal shock and excessive pressure at all times. Protection of the core shall receive first priority. The protective system shall not unnecessarily interfere with normal operation of the plant, and shall minimize unnecessary shutdowns. No single-component failure in the plant protective system shall prevent the necessary protective action from taking place or cause a plant shutdown. Interlocks shall be provided to prevent automatic and manual misoperation.

*An interval of time is called a steady-state interval if, during the preceding and succeeding 1/2 hr, steam temperatures do not vary by more than 25°F, pressure by more than 50 psi, and power by more than 1%.

1.3.8.5 Auxiliary Systems Segment – The auxiliary systems segment shall provide for makeup and drainage of primary and secondary main coolant, maintenance of the condition of the coolant and working fluid, maintenance of the condition of the inert gas over the reactor and intermediate coolant streams, maintenance of a suitable inert atmosphere in the reactor cavity and primary equipment cells, secondary component cooling and preheating during normal operations, secondary cooling of reactor under emergency conditions, and safe treatment and disposal of radioactive wastes. As noted previously, the reactor cover gas system shall include provisions to permit future modification for operation with vented fuel.

1.3.8.6 Electrical Segment – The electrical segment shall distribute, control, and meter the power generated by the power generation subsystem to the external power transmission system, and to the plant auxiliary electrical distribution systems. The electrical segment shall provide for startup of the entire system from a cold start, or from normal shutdown conditions. In addition, the electrical segment shall provide emergency, uninterruptible, and transient-free power as required. The electrical segment shall include necessary protective systems, such as grounding, lighting protection, and cathodic protection.

The in-plant electrical subsystem shall supply power for the control and operation of the plant at the following voltage levels:

4160 v, 3 phase, 60 cps, grounded wye

277/480 v, 3 phase, 4 wire, 60 cps

120/208 v, 3 phase, 4 wire, 60 cps

250 v dc

Motor Voltage: 250 hp and above, 4160 v – 3 phase;

1/2 to 250 hp, 480 v – 3 phase;

below 1/2 hp to 208, 3 phase

Any miscellaneous special voltages required for special equipment, devices, or circuits

The in-plant electrical subsystem shall include the necessary switchgear, transformers, motor control centers, panelboards, metering, controls, alarms, and related equipment. The in-plant electrical subsystem shall also include the communications systems. Emergency power shall be provided at all voltage

levels, as required. Uninterruptible and transient-free power shall be provided, as required.

1.3.8.7 Reactor Support Systems Segment – The reactor support systems segment shall provide on-site plant support functions, including:

- 1) In-core and ex-core fuel handling
- 2) Spent and fresh fuel storage, cooling, and shipping preparation
- 3) Reactor plant equipment decontamination and maintenance
- 4) Miscellaneous conventional power plant services
- 5) Reactor plant cranes
- 6) Reusable fuel shipping containers, exclusive of spent fuel shipping casks
- 7) Special tools and service equipment. Capability shall be provided for safe, reliable, and efficient handling and storage of all removable core elements, such as fuel elements, control rod absorber elements, neutron source, in-core instrumentation elements, moderator elements, reflector elements, and dummy core elements.

Handling of any elements which are within the reactor vessel shall be effected only while the reactor is shut down. Handling of any element outside the reactor vessel will be permitted while reactor is in operation, including fuel preparation for shipment to a separations plant. Storage facilities and capacities shall be determined on the basis of core element decay time storage and handling requirements.

Special consideration shall be given in the design to preventive maintenance of equipment while not in use, to assure its highest availability. To the extent practicable, equipment shall be accessible to permit operation and maintenance functions to be conducted with minimum interference to normal reactor operations.

1.3.8.8 Land, Structures, and Conventional Systems Segment – This segment shall comprise the fixed installations and associated services necessary to house the Plant and personnel. The site is defined as that area within the

control of the plant operator, and includes the exclusion area and approaches to the site.

The fixed installations shall:

- 1) Protect the plant equipment and personnel from damage during all credibly occurring natural phenomena
- 2) Provide an environment for reliable operation, maintenance, and storage of equipment, and for the comfort and safety of facility personnel
- 3) Control the release, within the acceptable limits, of any material or radiation that may be injurious to the health and safety of the populace.

The reactor containment building shall serve to protect the public from uncontrolled fission product release. The containment structure, including access openings and penetrations, shall accommodate, without damage, the maximum transients of pressure and temperature associated with credible accidents. The design leak rate from the containment structure shall be consistent with 10 CFR 100, based on analyses of the design basis accident (DBA). Provision shall be made to test the structure at equivalent design conditions at the completion of building construction. Temporary closures for penetrations, and protection for installed pressure-sensitive equipment, shall be provided for the construction test. Provision shall also be made for periodic leak rate testing of the completed structure at required pressures and temperatures.

If a ventilation stack is required, particular care shall be taken in its design and construction, to insure that it is as short and unobtrusive as possible and does not detract from the general appearance of the plant.

The plant shall be arranged on the site in such a manner as to permit orderly expansion by the addition of future plants of the same or larger capacity. Attention shall be given to arrangement of buildings to facilitate future expansion and common use of facilities by subsequent plants.

1.3.8.9 Seismic Design Criteria

1.3.8.9.1 Basic Requirement — Those systems and components which are essential to the prevention of accidents which could affect the public health and safety,

or to mitigation of their consequences, shall be designed, fabricated, and erected to withstand, without loss of the capability to protect the public, the additional forces that might be imposed by natural phenomena, such as earthquakes.

1.3.8.9.2 Seismic Classifications – Each system, component, and structure in the plant shall be reviewed and classified according to one of the following classifications:

1.3.8.9.2.1 Class I – Those systems, components, and structures, or parts thereof, whose failure might cause or increase the severity of a loss of coolant flow accident, or result in the uncontrolled release of excessive amounts of radioactivity; or which are vital to a proper shutdown of the plant and the removal of decay heat. Class I items shall be further assigned to one of the following subclassifications:

1.3.8.9.2.1.1 Class IA – Those Class I systems, components, and structures whose design, development, and testing should be conservative enough to permit installation and use at any site without requiring significant redesign and development for changes in seismic loads.

Examples of items that may fall into this Class IA include, but are not limited to, the following:

- 1) Reactor vessel, internals, and support
- 2) Core and core support structure
- 3) Loading face shield
- 4) Control rods and drive system, including equipment necessary for scram operation
- 5) Fuel assemblies
- 6) Primary sodium piping system, including valves
- 7) Primary sodium pumps
- 8) Intermediate heat exchanger
- 9) Decay heat removal systems
- 10) Reactor and process control and protective instrumentation

11) Fuel handling machines

12) Steam generators

1.3.8.9.2.1.2 Class IB — Those Class I systems, components, and structures which do not require a significant development program, and which have design requirements that are dependent on site conditions, thereby requiring specific final design for each site.

Examples of items that may fall into this Class IB include, but are not limited to, the following:

- 1) Reactor building and all containment structures
- 2) Fuel storage and handling facilities
- 3) Control room
- 4) Reactor building ventilation system
- 5) Inert gas systems
- 6) Secondary sodium system including valves and pumps
- 7) Pipe supports for primary and secondary sodium piping
- 8) Steam generator support structure
- 9) Emergency electrical power system

1.3.8.9.2.2 Class II — All other portions of the plant not falling into Class I.

1.3.8.9.3 Requirement for Class I Structures

1.3.8.9.3.1 Types of Earthquakes

Design Earthquake (DEQ): The maximum expected horizontal and vertical ground motions.

Maximum Hypothetical Earthquake (MHEQ): The maximum hypothetical horizontal and vertical ground motions.

1.3.8.9.3.2 Functional Requirements

Design Earthquake: All Class I structures shall be designed to withstand, without impairment of function, the loads and forces resulting from the DEQ ground motions, when combined with all other simultaneously occurring loads and forces.

Maximum Hypothetical Earthquake: All Class I structures shall be designed to withstand, without gross failure or loss of capability to achieve a safe shutdown of the plant, the loads and forces resulting from the MHEQ ground motions, when combined with all other simultaneously occurring loads and forces.

1.3.8.9.3.3 Design Seismic Motions

1.3.8.9.3.3.1 Class IA Structures

Design Earthquake: A horizontal ground or equipment support acceleration of 0.33 g, plus a vertical ground or equipment support acceleration of 0.22 g.

Maximum Hypothetical Earthquake: A horizontal ground or equipment support acceleration of 0.5 g, plus a vertical ground or equipment support acceleration of 0.33 g.

1.3.8.9.3.3.2 Class IB Structures:

Design Earthquake: The maximum probable horizontal ground motion at the specific plant site, plus a vertical ground motion that is $2/3$ the maximum horizontal ground motion. This maximum probable horizontal ground motion shall be determined by reviewing the geologic structure and seismic history of the plant site and the surrounding region, and utilizing the more severe of either the maximum recorded ground motion at the site or the maximum estimated ground motion at the site which would have resulted (but may not have been measured) from recorded earthquakes in the region.

Maximum Hypothetical Earthquake: The hypothetical maximum estimated horizontal ground motion at the site, plus a vertical ground motion that is $2/3$ the horizontal ground motion. The hypothetical maximum estimated horizontal ground motion shall be determined by reviewing all active faults in the region which could produce ground motion at the site, and locating the epicenter of the maximum intensity earthquake for each such fault at a location along the fault line with respect to the plant site which would produce the maximum ground motion at the site.

1.3.8.9.4 Seismic Forces and Stresses

1.3.8.9.4.1 Determination of Forces and Stresses – Stresses resulting from seismic loadings for both the DEQ and the MHEQ shall be determined for the more severe of the following methods of determining seismic forces:

- 1) Static lateral seismic forces in conformance with the Uniform Building Code (UBC) for the seismic zone established by the UBC for the plant location. Static vertical seismic forces that are 2/3 of the lateral forces.
- 2) Lateral seismic forces, as determined by response spectra methods for the maximum estimated horizontal ground or support acceleration. Vertical seismic forces, as determined by response spectra methods for a maximum vertical ground or support acceleration that is 2/3 the maximum horizontal ground acceleration.

1.3.8.9.4.2 Damping Coefficients – The following damping coefficients, expressed as percentage of critical damping, shall be used in response spectra calculations:

Welded Exposed Steel Structures	1%
Riveted or Bolted Exposed Steel Structures	2.5%
Reinforced Concrete Structures	5%

1.3.8.9.4.3 Combination of Seismic Stresses – The stresses due to horizontal earthquake motions and those due to vertical earthquake motions shall be added directly (in terms of absolute values).

1.3.8.9.4.4 Allowable Stresses – Allowable stress values for each Class I structure under the combined loading of earthquake plus all other simultaneously occurring loads and forces shall be selected to assure that the functional requirements previously described are satisfactory.

1.3.8.9.4.5 Basis for Determining Allowable Stresses – The determination of allowable stresses for each system, component, or structure shall take into consideration such factors as: materials of construction, basic design code, fabrication and quality assurance procedures, and inspectability and testability of the component or structure during the plant lifetime.

1.3.8.9.5 Requirements for Class II Structures

1.3.8.9.5.1 Minimum Requirements – All Class II structures shall be designed in accordance with the applicable provisions of the Uniform Building Code, latest edition, for the seismic zone established by the UBC for the specific plant location.

1.3.8.9.6 Local Codes – If local state or municipal government building codes are more stringent than the UBC, the more stringent local code shall govern.

1.3.8.9.7 Seismic Instrumentation – Strong motion accelerograph shall be installed at the plant site to accurately measure and record the ground accelerations associated with a strong earthquake. These data will be used to assess possible hidden structural damage.

1.3.8.10 Applicable Standards and Specification – The following standards and specifications form a part of the Design Requirements, to the extent specified. Further definition will be required in the area of standard engineering practices and procedures for design, fabrication, construction, quality control, inspection, proof testing, operation, maintenance, and repair of subsystems and components. The selection of specifications and standards used to satisfy these requirements shall be based, in the order of preference, on: (1) society, (2) federal, (3) state, (4) military, and (5) company sources.

- 1) ASME Boiler and Pressure Vessel Code, Section III
- 2) ASME Boiler and Pressure Vessel Code, Section VIII
- 3) ASME Power Boilers, Section I
- 4) ASA Code for Pressure Piping, B-31.1
- 5) Uniform Building Code
- 6) Walsh-Healy Act, as an Extension of the Uniform Building Code
- 7) Radioactivity Release – AECM-0324
- 8) National Electric Code
- 9) Pressure Piping Codes (ASA-B-31.1 Series)
- 10) National Plumbing Code (ASA-A-40.8)
- 11) National Fire Protection Association Standards (NFPA)

- 12) Fire Protection, per AECM-0552
- 13) 10 CFR 100 and 10 CFR 20
- 14) ANS 7.60, Proposed Standard for Leakage Rate Testing of Containment Structures for Nuclear Reactors
- 15) AEC Technical Safety Guide 7.5.1, Reactor Containment Leakage Testing and Surveillance Requirements
- 16) ASA N-6.2-1965, Safety Standard for Design, Fabrication, and Maintenance of Steel Containment Structure for Stationary Nuclear Power Reactors
- 17) American Institute of Steel Construction: Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings
- 18) American Iron and Steel Institute - Light Gage Steel Design - Manual
- 19) Steel Joist Institute: Standard Specification for Open Web Steel Joist Construction
- 20) Research Council on Riveted and Bolted Structural Joints: Specifications for Structural Joints Using ASTM A325 Bolts
- 21) American Welding Society
- 22) American Concrete Institute: Building Code Requirements for Reinforced Concrete, ACI 318
- 23) American Concrete Institute: Manual of Standard Practice of Detailing Reinforced Concrete Structures, ACI 315
- 24) National Bureau of Standards: American Standard Building Code Requirements for Minimum Design Loads in Buildings and Other Structures, B-58.1
- 25) American Association of Highway Officials
- 26) American Society of Heating, Refrigeration, and Air Conditioning Engineers

- 27) Safety Code for Cranes, Derricks, and Hoists, ASA B-30.2
- 28) National Electrical Manufacturing Association
- 29) Tubular Exchanger Manufacturers Association Standards
- 30) American Society for Testing Materials
- 31) Instrument Society of America
- 32) American Society of Mechanical Engineers (Applicable areas not specifically listed previously)

1.3.9 Software

Manuals and instructions in support of plant construction, plant testing, plant operation, plant maintenance, and personnel training shall be provided as follows:

- 1.3.9.1 Plant Construction Manual – This document shall provide construction sequence and interface control, and program the total construction phase.
- 1.3.9.2 Plant Testing Manual – This document shall identify and describe all testing conducted on-site, subsequent to completion of construction and prior to acceptance by the Customer. It shall include all pre-operational tests.
- 1.3.9.3 Power Plant Manual – This document shall provide a description of the plant, operating procedures for normal and emergency conditions, and routine test and calibration procedures.
- 1.3.9.4 Plant Maintenance Manual – This document shall define the scheduled maintenance program and the instruction necessary to accomplish this maintenance. It will also contain a trouble-shooting guide and fault isolation techniques. Operation and maintenance instructions, supplied with certain components and equipment, shall be included.
- 1.3.9.5 Health and Safety Manual – This document shall prescribe rules, regulations, and procedures which shall provide a general basis for working safely with radiation and/or radioactive materials.
- 1.3.9.6 Fuel Accountability Manual – This document shall specify the procedures for procurement, utilization, control, transfer, scrap recovery, losses, and return of fuel material to the government.

1.3.10 Spares

The need for spare parts and components shall be established by trade studies, which will weigh the cost of keeping spares on hand against the potential delays in plant operation if spares are not immediately available. Careful review and evaluation of equipment failure rate data and component procurement times will form key inputs for these trade studies.

2.0 DETAILED DESCRIPTION

(The detailed description is covered in Section IV
of AI-68-MEMO-37, Vol I)

3.0 PRINCIPLES OF OPERATION

The plant, because of its low incremental power costs, is expected to operate as a base load installation during a significant portion of its lifetime. The plant, however, will be designed for load following over the range of 25% power to 100%.

3.1 Normal Operation

The major activities involved in normal operation are as follows.

3.1.1 Power Demand

In general, the plant will be operated in the automatic load-following mode. Electrical generation will be adjusted to meet the demand requested by the dispatcher, or the station will automatically change in load as directed by the automatic frequency control. Automatic operation will be possible over a load range from 25 to 110% of rated power, with manual or semi-automatic control option at any power level. The automatic control system will control the power output of the steam generators by varying the flow of primary sodium, secondary sodium, and boiler feedwater in response to power demand. Reactor power will be controlled (by control rod movement) from primary sodium flow. In addition to these primary controls, there will be trim controls on reactor hot and cold leg temperature. The operators will maintain a continuous surveillance of important plant variables, to ascertain the proper operation of the controls and log the data required for plant records. However, deviation alarms on important control points will require the operators to make manual adjustments to the affected system until the deviation is corrected, the controller realigned, and the system returned to fully automatic control.

An analytical program will monitor key areas in the plant to detect abnormal conditions which might affect plant availability or safety. An example is the reactivity normalcy program, which will compare the actual reactivity with the predicted reactivity, as an aid to the plant operator in detecting possible core performance abnormalities.

3.1.2 Shim Rods Burnup Adjustment

Periodically, during the nominal six-month (between refuelings) operating cycle, shim rod positions will be adjusted to compensate for the core reactivity decrease due to fuel burnup.

3.1.3 Sodium Purification

Primary and secondary sodium will be cold trapped, as required, to remove oxygen, hydrogen, and some fission products.

3.1.4 Sodium Purity Monitoring

Periodic measurements of primary and secondary sodium plugging temperature will be made. These plugging temperature measurements will yield information regarding sodium system oxygen concentration. If the oxygen level is too high, cold trapping will be performed as required.

3.1.5 Cover Gas Purity

The cover gas purification system will be operated, as required, to maintain acceptable levels of fission product activity (primarily xenon and krypton) in the reactor cover gas.

3.1.6 Cover Gas Monitoring

In addition to the measurements of activity level of fission products, made as part of the cover gas purification process, the cover gas will periodically be monitored for other gaseous impurities, such as oxygen, nitrogen, water vapor, and carbon dioxide.

3.1.7 Containment Dome Gas Monitor

A nitrogen atmosphere will be maintained in the containment dome, and the gas will be monitored for fission products.

3.1.8 Pipe Cell Inert Atmosphere

A nitrogen atmosphere will be continuously maintained in the cells with equipment containing primary sodium.

3.1.9 Cell Atmosphere Monitoring

The nitrogen atmosphere in the primary equipment cells will be monitored continuously and automatically for the presence of oxygen, water vapor, and radioactivity.

3.1.10 Liquid Waste Disposal

Liquid wastes, if sufficiently low in activity level, will be periodically released to the cooling water effluent or other suitable discharge stream. If

the activity level is above the discharge level, the liquid waste will be transferred to a tank truck for off-site disposal.

3.1.11 Liquid Effluent Activity Monitoring

Prior to release of any liquid waste, it will be monitored for activity level by sampling. During release, the activity of the effluent will be continuously monitored and recorded by automatic instrumentation. If the activity rises above a set level, the release will automatically cease.

3.1.12 Radioactive Gas Disposal

Low-activity gas will be released directly to the atmosphere. Higher-activity gas will be compressed, collected in decay tanks, and allowed to decay before being released at a controlled rate.

3.1.13 Gaseous Effluent Monitoring

Gas samples will be obtained and checked for activity levels before release from the decay tanks. Continuous activity monitors, located in the radioactive gas collection piping and in the release duct, will shut off release if activity exceeds a pre-set value.

3.1.14 Decayed Fuel Shipping

Spent fuel will be shipped off site for reprocessing.

3.1.15 New Fuel in Ex-Vessel Storage

New fuel elements arriving on site will be placed in ex-vessel storage.

3.1.16 Fuel Handling Machine Checkout

The fuel handling machine will be thoroughly checked out and repaired, as necessary, to assure high probability of continuous trouble-free operation during the next shutdown.

3.1.17 On-Line Maintenance

As much as possible of the plant will be maintainable while the reactor is on the line (e.g., by use of multiple units). This equipment will be checked and repaired, as necessary, on a scheduled basis. Of particular importance will be the various redundant units making up the protective system.

3.1.18 Periodic Safety Tests

Where possible, tests to verify the continuing safety of the plant will be performed during reactor operation. An example of such a test is the measurement of the reactor power coefficient by making small step changes in power.

3.2 Normal Startup (After Fuel Handling)

The status of the plant systems after refueling and just prior to plant startup will be as follows.

3.2.1 Main Heat Transfer Systems

Sodium will be flowing through all main heat transfer loops, and the sodium temperature will be nearly isothermal at 600°F.

3.2.2 Steam Generators

Feedwater will be circulating through the evaporator to the flash tank. Evaporator pressure and feedwater flow will be adjusted to maintain a 600°F sodium temperature. Steam from the flash tank will be circulating through the superheater and reheater. The majority of the steam exiting from the superheater and reheater modules will undergo a pressure reduction and be admitted to the condenser; the remainder of the steam will be used to maintain the temperature of the main steam and reheat steam lines to and from the turbine.

3.2.3 Steam Plant

Steam from the auxiliary boilers will be flowing to the flash tank. Steam from the flash tank will be flowing to the feedwater heaters, deaerator, and boiler feed pump turbine. Feedwater will be circulating through the low-pressure heaters, deaerator, demineralizers, and high-pressure heaters. The boiler feed pump and the decay heat condensate pump will be providing the pumping power for circulating the feedwater.

The cold startup from this point to 100% power is similar to the initial startup outlined in Section 6.0, except for the following: (1) the starting temperature of the circulating sodium will be higher, (2) the testing conducted during the initial startup will be omitted, (3) the steam pressures and temperature will be higher, and (4) the turbine may be started at ~5% power.

3.3 Startup After Scram

3.3.1 Reactor Trip

Reactor trip is a rapid shutdown of the reactor plant. In the power range, measured high flux, measured high temperature, low flow, low reactor sodium level, or manual initiation will cause trip. When trip occurs, the safety rods and the shim-safety rods will be released, power to the pump drives will be cut off, and the throttle valves in the primary and secondary heat transfer circuits will start driving shut. Turbine trip will also take place. Upon turbine trip, the block and intercept valve will close, the steam dump valves on the superheaters and reheaters will open and dump to the condenser. The steam dump valves and piping will handle 18% of full-power steam flow. Excess steam will be exhausted to the atmosphere through the power relief valves.

Following reactor trip, the pumps will be restarted at minimum speed after a small delay, and the throttle valves will control the natural convection sodium flow in the primary and secondary circuits to control the decay of reactor outlet temperature to $\sim 100^\circ\text{F/hr}$. The sodium flow will be continued through the superheaters and evaporators, but the reheater will be valved off. Feed-water, heated in the topping heater by steam from the superheaters, will continue to be pumped to the evaporator. After the main turbine is tripped, feed-water will be supplied by the decay heat condensate pump and the boiler feed pump. Approximately 30 min to 1 hr after the scram, the effluent from the boiler will be directed to the flash tank. The standby auxiliary boiler will be brought on line, and steam from the boiler will be fed into the flash tank. Steam from the flash tank will be circulated through the superheater and reheater. Steam exiting from the superheater and reheater will be directed to the condenser. At this point, the plant will be lined up, as outlined in Section 3.1. When the main heat transfer loops are nearly isothermal, which will take ~ 4 hr, the reactor will be ready for restart, using the normal startup procedure.

During the temperature decay period, the cause of the trip will be determined, control rods driven to their lower limit, all startup checks completed, and critical rod position calculated.

3.3.2 Turbine Trip

A trip of the turbine-generator does not require reactor trip, although it is possible that the reactor will trip in the ensuing upset of the electrical system. If the reactor does not trip, the automatic control system will begin lowering the reactor power automatically at the maximum controllable rate. Upon turbine trip, the intercept valve will close, the steam dump valves on the superheaters and reheaters will open and dump to the condenser. The steam dump valves and piping will handle 18% of steam flow. About 82% of the steam flow will be exhausted to the atmosphere through the power relief valves. Reactor power will continue to drop until it matches the capability of the steam dump system. The reactor can then continue to be shut down, or the turbine-generator may be restarted and placed on the line. The steam bypass around the turbine to the condenser will be gradually reduced, if the turbine is to be restarted, until all the steam is flowing to the turbine. The plant will then be brought up to 100% power, as outlined in the rise-to-power paragraphs of Section 6.0.

3.4 Normal Shutdown

Reactor plant shutdown is normally for the purpose of refueling, turbine maintenance, and maintenance on other plant components which can only be worked on during shutdown. The major activities for normal shutdown will be as follows.

3.4.1 Reactor Power Reduction

The reactor power will be reduced by driving in the control rods, which lowers the neutron flux. The sodium flow and reactor ΔT will also be reduced until reactor power is lowered to about 5 to 10%. At 5 to 10% power, depending upon the turbine design, the generator will be tripped from area load frequency control, and the turbine electrical load will be reduced until the underspeed release trips the turbine. Upon turbine trip, the intercept valves will close, the steam dump valves on the superheaters and reheaters will open, and the steam will be reduced in pressure and exhausted to the condenser. The sodium flow through the reheaters will be reduced to zero, and the reheaters will be isolated.

Feedwater, heated in the topping heater by steam from the superheaters, will continue to be pumped to the evaporator. The feedwater will be supplied by the decay heat condensate pump and the boiler feed pump. Power for

the electric drive on the decay heat condensate pump will be provided by the emergency power bus.

3.4.2 Reactor Outlet Temperature Reduction

The reactor outlet temperature will be reduced at a controlled rate of about 100°F/hr while reducing power. This is accomplished by keeping the sodium flow constant while lowering neutron flux by driving in the control rods. This will result in a gradual reduction of steam flow to the steam dump. The reactor will be tripped when there is no longer a significant temperature rise across the core. At ~1% power, the effluent from the boiler will be directed to the flash tank. The standby auxiliary boiler will be brought on line, and steam from the boiler will be fed into the flash tank. Steam from the flash tank will be circulated through the superheater and reheater. Steam exiting from the superheater and reheater will be directed to the condenser.

3.5 One-Circuit Shutdown

Loss of sodium or feedwater flow will not cause a reactor trip, if only one of the three main heat transfer circuits is affected. Instead, only the affected circuit (primary loop, secondary loop, and feedwater/steam loop) will be taken out of service. In a scheduled shutdown of one circuit, the flow of primary and secondary sodium and the flow of feedwater to the evaporator of the circuit to be shut down will be gradually reduced. At the same time, the power demand from the turbine-generator will be reduced sufficiently to overcompensate for the loss of steam from the loop being shut down. When the flow of sodium in the affected loop reaches about 15%, the primary and secondary sodium pumps will be shut down, and the block valves in the primary and secondary loops and the feedwater systems will be closed, isolating the loop. At the same time, the power demand will be reduced to about 60%. When the shutdown loop is isolated, the power demand can be reset to about 70%, since this will be approximately the power capability of the remaining two loops. Temperatures in the shutdown circuit will gradually decay. If the circuit can be restored to service before the hot leg temperature drops more than about 100°F, then a reactor shutdown, prior to returning the third loop to service, will not be required. However, if the circuit is out of service for a period of time permitting its hot leg temperature to drop more than 100°F, it will be necessary to shut

the reactor down, allow its temperature to drop to match that of the shut down circuit, restore the shut down circuit to service, and then restart the reactor. The startup procedure will be the same as outlined in Section 3.2. When one loop is out of service, faults in either one of the two operable circuits will cause complete reactor shutdown.

4.0 SAFETY PRECAUTIONS

4.1 Hazards

The major hazards which could arise, and which should be the subject of formal casualty studies, include:

- 1) Fuel pin cladding failure
- 2) Inadequate sodium flow to one or several fuel elements
- 3) A large unscheduled reactivity insertion
- 4) Low reactor coolant level
- 5) Primary piping system leakage
- 6) Heat sink failure
- 7) Inadequate total core coolant flow
- 8) Protective system failure
- 9) Fuel handling machine loss-of-cooling
- 10) Excessive radiological release in containment building.

4.2 Safety Precautions

The safety precautions for the plant will be governed primarily by the "General Design Criteria for Nuclear Power Construction Permits," 10 CFR 50, and associated federal regulations. Appropriate state, local government, insurance, and utility regulations and practices must also be observed. The safety considerations particular to fast reactors and the use of sodium as a coolant must be taken into account, in design and in plant operation.

5.0 MAINTENANCE

The overall maintenance program will be based on the plant availability goals set forth in Section 1.3.2. That is, the maintenance frequency and type will be planned to permit achievement, by each major plant segment, of the availability goals specified. In some cases, this will require stocking of complete spare components (e.g., main heat transfer pump internals). In other cases, it will require the capability for on-line maintenance (e.g., individual channels of the protective system). In still other cases, inspection and required maintenance will be performed on a periodic basis with the plant shut down (e.g., the steam turbine).

Based on the current conceptual design, the maintenance methods and policies for the various major plant system and components are as follows.

5.1 Reactor System

5.1.1 Vessel and Internals

The reactor vessel is not expected to require maintenance during its lifetime. The vessel internal permanent structure (such as the grid plates) will be inspectable and maintainable by means of remote equipment. Provision will be made for complete draining of the reactor vessel. The grid plates will be removable.

5.1.2 Fuel and Blanket Assemblies

Fuel and blanket assemblies will be removable and replaceable, using the fuel handling equipment. Sufficient storage space will be supplied to permit complete unloading of the entire reactor, if necessary. Fuel and blanket assemblies will be capable of remote disassembly and inspection, either in an on-site or off-site hot cell. Provisions will be made for handling a reasonable range of abnormal conditions, such as stuck or damaged fuel elements.

5.1.3 Control Rods

Spare control rod drive systems and poison elements will be supplied, as required, to meet the reactivity control system reliability requirements. Control rod assemblies will be removable and replaceable with the fuel handling equipment.

5.1.4 Loading Face Shield

The rotating loading face shields will incorporate replaceable bearings and seals, for which spares will be provided. Capability will be provided for monitoring seal leakage at any time.

5.2 Heat Transfer System

5.2.1 Primary Block Valves

Failure rate for these valves is expected to be very low, since the valves will be closed only in the event that it is necessary to isolate one of the heat transfer circuits. Valve operator drives will be maintainable from the reactor building floor during plant operation. Repairs to the valves and the portion of the drive train in the shielded vaults will require a complete reactor shutdown, and 10 to 15 days for decay of sodium activity. Removal of a block valve or replacement of the gates will require draining the primary loop and the reactor vessel below the level of the valve. Decay heat removal will be provided by the other loops. The valves will be checked for operability during each scheduled reactor shutdown.

5.2.2 Primary Throttle Valve

Valve operator drives will be maintainable from the reactor building floor during plant operation. Repairs to this valve will also require entry to the primary gallery, after 10 to 15 days of sodium activity decay. The decay can take place with the reactor in operation on the remaining two heat transfer circuits and with the block valves on the affected loop closed. Actual repairs to the valve will require a complete reactor shutdown. The reactor vessel sodium level will not have to be lowered, because the throttle valve is isolated from the reactor by the block valves.

5.2.3 Intermediate Heat Exchanger (IHX)

It is expected that, if repairs are required to the IHX, they will either be repairs to a leaking tube or a repair of a possible leak in the shell. A leak in the shell will result in secondary sodium leaking into the primary gallery (but no sodium fire, because of the nitrogen atmosphere in the vault), and will require a shutdown of the affected loop, a 10- to 15-day decay period, and then a complete reactor shutdown while the gallery is entered, the solidified

sodium cleaned up, and the shell repaired. If significant quantities of fission products are present in the primary system, it may be necessary to isolate the IHX and subject the primary side to chemical cleaning, to reduce the radiation to working levels.

Tube repairs may be accomplished in a number of ways, depending on the final design of the unit. If the tube bundle is removable, it can be removed and replaced with a spare bundle, using either a cask or a plastic bag, after allowing 10 to 15 days for sodium decay. The reactor would continue to be operated at part load on the remaining two heat transfer circuits, while waiting for decay in the third circuit. If the tube bundle is not removable, but the upper tube sheet is accessible without entering the primary gallery, a leaking tube can be plugged in place, using remote leak detecting and tube welding equipment. The reactor would continue to be operated at part load in this case also. Finally, if the unit requires entry into the vault to repair a leaking tube, the operation would be similar to that required for repair to the shell. The leaking tube would be plugged in place, with the reactor shut down.

5.2.4 Primary Pump and Drive System

Repairs to the pump drive system and accessible pump subcomponents, such as the shaft gas seal, will be made with the pump shut down and with the reactor operating at part load on the remaining two heat transfer circuits. Repairs requiring removal of the pump internals will be accomplished by removing the internals as a unit, using a shielded cask. Spare pump internals will then be installed. The reactor will be operated during this effort, with the block valves shut and the affected primary loop drained.

5.2.5 Primary Piping and Pump Casing

A leak in the primary piping or pump casing has a very low probability of occurrence, and will not be accompanied by a sodium fire, because of the nitrogen atmosphere in the galleries. Such a leak can be repaired by closing the block valves on the leaking loop, draining it, allowing the sodium to decay for 10 to 15 days while operating the reactor on the remaining two circuits, then shutting down and entering the gallery, cleaning up the spilled sodium (which will be solidified), and repairing the leak. Flushing may be required to reduce the level of fission product activity.

5.2.6 Secondary Block and Throttle Valves

Secondary block and throttle valves are accessible at all times. Repair will be accomplished with the reactor operating at part load on the remaining two heat transfer circuits.

5.2.7 Secondary Pump and Drive System

The secondary pump will be repaired in the same manner as the primary pump, except that a shielded cask will not be required for pump removal and replacement with spare internals. Instead, a plastic bag will be used to preserve the inert atmosphere in the secondary system. The reactor will be operated at part load on the remaining two heat transfer circuits while the repairs are being made.

5.2.8 Steam Generators

The steam generators are expected to be highly reliable, due to their design simplicity and the inspectability of the tube-tubesheet welds. However, tube failures, permitting water to come into contact with secondary sodium, may take place. The affected secondary loop will be shut down, drained on both the sodium and water sides, and the module containing the leaking tube cut out and replaced with a spare module. The secondary loop will then be refilled, and the sodium circulated through the system cold trap as necessary to reduce oxygen and hydrogen contamination to an acceptable level. During these repairs, the reactor will be operated at part load on the remaining two circuits.

5.2.9 Secondary Piping, Expansion Tank, and Steam Generator Shells

The piping, expansion tank, and steam generator shells are designed for long life and high integrity. In the unlikely event of a sodium leak, the affected system will immediately be shut down, while the reactor continues operation on the remaining two heat transfer loops. The system will be drained as rapidly as possible, and the probable fire extinguished with calcium carbonate or other "dry" fire fighting material.

5.3 Turbine-Generator

The turbine-generator and its associated equipment (condenser, condensate pump, feedwater heaters, feedwater pumps, etc.) are conventional equipment, and will be maintained in accordance with standard utility practice.

The schedule for inspection of the turbine stages will be coordinated with the reactor refueling schedule to make optimum use of the time during which the entire reactor plant must be shut down. For example, one stage of the turbine may be inspected and repaired, as necessary, during an annual two-week shutdown. For about six days, running concurrently with the turbine inspection and maintenance, fuel handling will be performed.

5.4 Control and Instrumentation

Adjustment and on-line maintenance of control and instrumentation equipment is expected to absorb a large percentage of the total maintenance effort.

5.4.1 Plant Control System

Except for the last control element in a chain (for example, a control valve), the plant control system will be capable of being maintained with the reactor at power. Individual portions of the control system will be placed on manual control while adjustments and repairs are being made.

5.4.2 Plant Protective System

By utilizing two out of three coincidence channels throughout the plant protective system, it will be possible to maintain any channel, replacing components with spares, while the reactor is on the line.

5.4.3 Nuclear Instrumentation

All nuclear instrumentation channels will be triplicated, making it possible to perform maintenance on one channel without shutting the reactor down.

5.4.4 Radiation Monitoring

In general, individual components of the radiation monitoring system will be repaired while performing "hand" monitoring, using substitute instruments. Reactor shutdown will not be required.

5.5 Auxiliary Systems

These systems include the sodium purification and supply system, the inert gas supply systems, the radioactive waste gas system, the liquid waste system, the heating and ventilating system, the cooling water and kerosene system, and the preheat system.

5.5.1 Coolant Supply and Treatment System

Two redundant primary purification subsystems are included in the plant design. These subsystems include a cold trap, a hot trap, sodium impurity measuring equipment (e.g., a plugging meter), pumps, valves, and piping. One entire subsystem may be removed from service for replacement of a cold trap or repair of a component, while the other subsystem is in use. Maintenance will be performed by allowing 10 to 15 days for decay, and then entering the purification system vault. Cold traps containing significant quantities of fission products will be remotely disconnected from the piping system and placed into a disposal cask. The secondary purification subsystems, one for each secondary loop, will be nonradioactive. Any one of these systems can be removed from service for a short period of time without shutting down the associated secondary loop.

5.5.2 Inert Gas Supply Systems

These systems must be in essentially continuous operation. Manual bypass lines are included around all control elements to permit maintenance or replacement of the control element without interruption of service.

5.5.3 Radioactive Waste Gas System

This system, consisting of pressure controllers, compressors, and decay tanks, will normally be accessible on a limited time basis. Pressure controllers will generally have manual bypass lines to permit maintenance. Compressors and decay tanks will be redundant, to permit maintenance on one unit while the other is running.

5.5.4 Liquid Waste System

In general, operation of the liquid waste system can be temporarily interrupted for repairs without affecting reactor operation.

5.5.5 Heating and Ventilating System

Maintenance is performed by shutting down one of two redundant units. For example, two 100% capacity fans are provided for containment building ventilation and for cooling of the various galleries.

5.5.6 Cooling Water and Kerosene System

Two redundant pumps and two redundant heat exchangers are provided for both the kerosene system and the closed loop cooling water system, to permit maintenance without system shutdown.

5.5.7 Preheat System

Redundant heaters, operating at partial voltage, are supplied on primary piping. Maintenance is expected to be very infrequent, but would require shutdown and decay of the affected area. In the secondary system, heaters can be replaced by shutting down the affected loop while running the reactor at part load on the remaining two circuits. The remaining parts of the preheat system are on vessels and auxiliary lines not normally used during reactor operation, and therefore required maintenance can be accomplished without reactor shutdown.

5.6 Electrical System

The electrical system is conventional, and is maintained in accordance with standard utility practice.

5.7 Support Equipment

Support equipment consists primarily of the in-core and ex-core fuel handling machines, and other machines, such as the plug handling cask, sweep arm, and shipping cask. This equipment is required to be used periodically during refueling operations. When not in use, the equipment will be thoroughly checked, and repairs made as necessary to assure trouble-free operation. In general, these repairs can be made by direct contact maintenance. Decontamination of machine parts occasionally may be required.

5.8 Site and Structures

The containment building periodically will be checked for leakage rate. This can be accomplished during routine reactor shutdowns. Other building maintenance will be in accordance with the established practice of the utility.

6.0 TEST REQUIREMENTS

6.1 Initial Startup

Testing is required to determine that the plant meets the safety, design, and performance requirements.

6.1.1 Precritical Period

During the precritical phase of plant startup, all plant systems will be checked out, the sodium systems filled, the fuel handling equipment checked out, and the reactor readied for fuel loading. As each plant system is completed and tested, it will be turned over to the operating crew for functional checkout. During the checkout, instrumentation will be calibrated.

6.1.2 Startup Testing Program

6.1.2.1 Low Power Phase – The control rods and special flux detectors will be installed, and fuel will be loaded in increments to achieve the operational core loading. The critical loading will be checked. With the core loaded, core pressure drop will be measured as a function of flow. Control rods will be calibrated, and "zero power" reactivity coefficients will be measured. These will include the isothermal temperature coefficient, sodium flow coefficient, cover gas pressure coefficient, sodium level coefficient, and sodium void coefficient. A check of axial and radial flux profiles will be made, using the in-core flux detectors.

6.1.2.2 Rise to Power – During this phase, the plant power level will be increased in steps. At each step, selected tests will be performed. They will be chosen, as appropriate, from the following:

- 1) Power coefficient measurement
- 2) Core and equipment temperature distribution measurements
- 3) Overall plant heat balance
- 4) Plant radiation survey
- 5) Control rod calibration
- 6) Plant control system performance check

- 7) Steam generator performance check
- 8) Vibration checks
- 9) Reactor trip
- 10) Turbine trip
- 11) One-circuit shutdown
- 12) Reactor stability (transfer function) measurements
- 13) Ex-core fuel handling check

A power level increase of a factor of two is expected to be acceptable between major "plateaus" for testing. The schedule of power and temperature increases therefore will be approximately as follows:

Power Level (%)	Reactor ΔT ($^{\circ}F$)	Heat Sink
5	15	Steam dump
10	30	Steam dump
15	45	Steam dump and T.G.
15	90	T.G.
30	180	T.G.
60	300	T.G.
100	300	T.G.

T.G. = Turbine-Generator

This schedule, of course, is not intended to imply that large steps in power are taken without frequent pauses to check plant operating data. On the contrary, it is expected that the plant would be leveled about every 5 to 10%, to check the status of all equipment and ascertain plant normalcy before proceeding.

The procedure for increasing from low power to full power will be as follows:

6.1.2.2.1 Sodium Circulation - Sodium circulation, at 25 to 100% of full flow, will be established in all operable sodium loops. The reason for having sodium flow at a relatively high value will be:

- 1) To permit the protective system to operate in an optimum manner to prevent possible fuel overheating
- 2) To minimize the possibility of sodium boiling in a fuel channel
- 3) To avoid plant operation in a mode which can readily result in excessive temperature transients
- 4) To avoid a possible local positive power coefficient.

6.1.2.2.2 Pre-Startup Checks – These activities will consist primarily of instrumentation checks, to verify that required control and protective instrumentation is operable and that the setpoints are as specified. In addition, key plant equipment, particularly that required for emergency or abnormal plant conditions, will be checked out. An example of such a piece of equipment is the emergency diesel-generator.

6.1.2.2.3 Steam Generator – With the steam generators filled with steam on the tube side, and maintained at 350°F by circulation of the secondary sodium, the system will then be ready for increasing power.

Feedwater, circulated through the high-pressure heaters, will be bypassed to the flash tank until the feedwater temperature is 350°F. Heating of the feedwater will be accomplished in the high-pressure heater with steam from the auxiliary boiler, which is supplied to the flash tank. Steam from the flash tank will also be used to drive the turbine driven boiler feed pump. During this phase, feedwater flow will be established through the demineralizer, the de-aerator, and the low-pressure feedwater heaters. This flow will be provided by the electric driven decay heat condensate pump. Feedwater purity will be established.

Feedwater at 350°F will then be admitted to the evaporator modules; and, upon exiting, it will be fed to the flash tank. Steam from the flash tank will then be admitted to the bottom of the superheater, the reheater modules, and the high-pressure feedwater heater. A portion of the steam from the superheaters and reheaters will be used to heat the main and reheat steam lines; the remainder will be directed to the condenser.

The system flow condition will then be sodium circulating through the steam generators at 350°F, feedwater circulating through the evaporators at 350°F, and steam circulating through the superheater and reheaters.

6.1.2.2.4 Startup Interlocks — It will be necessary for certain plant conditions to exist, in order to permit the safety rod drives to be actuated. Examples of such conditions are: neutron count rate greater than some selected minimum value, no scram or setback conditions in existence, all shim and safety rods at their lower limit, etc. The operator must take whatever action is necessary to satisfy these conditions. In general, operating philosophy will be that no interlocks will be bypassed.

6.1.2.2.5 Control Rods — After calculating the expected critical position for all shim rods, the safety rods will be withdrawn while carefully watching for abnormal indications of neutron flux increase. The shim rods will then be withdrawn manually until the reactor is just critical, but no temperature increase has been observed. The neutron flux rise will be leveled, and the rod positions will be checked against the prediction. If reactivity is found to be within a specified range, as compared to that predicted, startup will continue. If not, startup will be delayed until the discrepancy is resolved by plant supervision.

6.1.2.2.6 Increase in Power and Temperature — The flow control system in the primary and secondary loops will be adjusted for full flow. Reactor power will then be increased manually until neutron flux is about 1 to 2% of full power. At this point, it will be possible to place the reactor on automatic flux control and raise the reactor outlet temperature at the design rate (nominally, 100°F/hr). During this rise in power, steam quality will be improving, and steam will continue to be dumped to the condenser.

Initially, the systems will be nearly isothermal at reactor inlet temperature. As reactor sodium outlet temperature is raised, secondary system sodium temperature will increase, and boiling will be induced in the evaporator. The two-phase mixture exiting from the evaporator will be directed to the flash tank, and the steam exiting from the superheaters and reheater will be admitted to the condenser or will be used to heat the main and reheat steam lines. The steam pressure in the evaporator will be adjusted to maintain a constant reactor inlet temperature. The reactor power will then be further raised, and the sodium flow adjusted to increase reactor ΔT at the rate of about 100°F/hr.

6.1.2.2.7 Turbine Generator — As the reactor power and sodium system temperature increases, the feedwater pressure and temperature will be increased to the desired value. The system temperature and pressure will be increased until the reactor is at about 15% power. At that point, the turbine intercept valves will be opened, and steam will be slowly admitted to the H. P. turbine. Simultaneously, steam from the flash tank to the superheaters and reheaters will be blocked closed. The steam exiting from the H. P. turbine will go to the reheater and return to the I. P. turbine. In this manner, the turbine will be rolled and brought up to a steady-state condition. The turbine will then be synchronized with line frequency and placed on the line. As the turbine is loaded to about 15% power, the steam flow from the superheaters to the condensers will be gradually reduced, and the steam flow to the turbine increased, until the turbine is loaded to about 15% power. At this point, steam from the evaporator modules will be gradually admitted directly to the superheaters, while the evaporator steam flow to the flash tank and the steam flow from the flash tank to the superheaters are reduced to zero. The steam dump system will then be placed on standby. The nuclear plant can now be placed on automatic load-following control.

6.1.2.2.8 Power Increase — The turbine-generator plant operator will increase power, as desired, by means of the turbine governor control, in increments as outlined before, stopping at each increment to conduct performance tests until 100% of power is reached. When the tests are completed and the plant is accepted, it is adjusted automatically to follow power demand or on automatic frequency control.

During the startup to power, the auxiliary systems, as required, will be placed on line.

IV. PLANT DESCRIPTION

The nuclear plant described on the following pages is a 1000-Mwe (net) electrical plant consisting of a sodium-cooled fast-neutron-spectrum reactor, its associated refueling equipment, a sodium heat-transfer system, sodium-heated steam generators, and a high-efficiency turbine-generator plant. The reactor, primary coolant system, and refueling system are housed in a reinforced concrete containment building, while the secondary coolant system, the steam generators, and the turbine generator are housed in a steel framed building covered with corrugated aluminum. The plant with its associated auxiliary systems and buildings can be located on a typical site with a 2000-ft exclusion radius.

The plant design given herein represents the first large LMFBR to be placed in operation following the operation of a demonstration plant of the same basic design. As such, its design and technology are based on AI's current demonstration plant design program.

A. PLANT ARRANGEMENT AND STRUCTURES

The conceptual site plot plan for the 1000-Mwe plant is shown in Figure IV-1 and the overall plant and building arrangement in IV-2. The orientation of the plant is representative of a typical layout for a multi-unit site and shows general features which might be expected for a normal site. Local topography, soil and climatic conditions, and transportation facilities of any specific selected site will influence this arrangement.

The plant comprises the following principal buildings: The reactor building, secondary-sodium and steam-generator building, turbine-generator building, plant control area, fuel-handling and radioactive-waste building, and the administration and control area. The arrangement, location, and relationship of the buildings are established to maintain the natural flow pattern of the system from the reactor and primary sodium system, through the secondary-sodium system and steam generator, to the turbine-generator. The arrangement is as closely coupled as practicable to minimize piping runs and operating and construction costs.

The plant control area is centrally located adjacent to both the reactor building and turbine-generator building. The close-in but separate building housing the control room minimizes power and control wiring while still permitting full-time occupancy during all normal and emergency plant conditions.

The basic plant arrangement is such that a second unit could be added at a later date with minimum cost and disturbance to the existing unit and with maximum use of existing facilities which could serve both units.

Railroad access will be provided to the turbine-generator building, switchyard, sodium unloading area, and fuel-handling building.

1. Reactor Building

The reactor building, Figure IV -3, houses the reactor and fuel-handling equipment and all portions of the primary-sodium systems, up to and including the intermediate heat exchangers. The building is a conventional reinforced concrete structure having a configuration of a right circular cylinder with a flat base and hemispherical head. Nominal dimensions are 176-ft ID, 105-ft inside height of cylindrical section, and 88-ft inside radius of the hemispherical head, for an overall inside height of 193 ft. The operating floor (top of primary heat-transfer cells) is at grade and the top of the base slab is 55 ft below grade. The reactor vessel, primary-sodium heat-transfer loops, and the drain tanks are located within reinforced concrete cells inside the building below the operating floor. The top shield covering the reactor vessel, the control-rod drives, and the fuel-handling equipment are located on the operating floor.

The reinforced concrete structures serve both as pressure containers and biological shields. The cylinder and dome sections of the building are 2-ft-thick and the base slab is 5-ft-thick. Walls and slabs of the primary cells range from 3- to 6-ft thick depending on shielding requirements and structural spans. Shielding is provided to permit full-time occupancy within the reactor building during normal plant operation. Removable concrete plugs in the reactor room floor provide access to the primary-sodium equipment cells. All cells containing primary-sodium equipment within the building are maintained under an inert atmosphere to prevent fire in the event of a sodium leak

Figure IV-1. Plot Plan
(Dwg. N09691001)

Figure IV-2. Plant Arrangement
(Dwg. N09690007)

Figure IV-3. Reactor Building
(Dwg. N09696006)

and to reduce the consequences of the DBA. The primary-sodium equipment cells are cooled by internal forced-circulation of the inert atmosphere over the cooling coils. Motor drives for the primary pumps and throttle valves are mounted on the operating floor. Space is provided either above or adjacent to the purification (cold trap) and drain tank cells for operation of small valves and local gage boards.

Containment of radioactive material is accomplished by double-containment barriers. The first barrier consists of (1) a 1/4-in.-thick welded-steel liner which separately encloses the reactor cavity, primary heat-transfer cells, and the drain tank cell, and (2) a carbon steel containment dome over the reactor top-shield. The plate liner is attached to and backed up by the reinforced concrete structure which resists the pressure loadings, and the containment dome is anchored to the cavity structure. The second barrier is a 3/8-in.-thick welded steel liner on the inside surface of the overall building envelope. Penetrations through these barriers for access, services, and process piping are designed to maintain the required pressure and leakage integrity.

The building is provided with a heating, ventilating, and air conditioning system, and an inert-gas system to provide a controlled and conditioned environment to the various air and inert-gas atmospheric areas within the building.

Personnel and small equipment access into the building will be controlled through airlocks that maintain the pressure and leakage integrity of the containment. Fuel and other equipment access is through an equipment airlock leading into the Fuel-Handling and Radioactive Waste Building. In addition a large-diameter equipment access hatch is provided for use during both initial construction and major maintenance or modification operations when the plant is shut down. Full operating floor crane coverage is provided by a polar crane installed within the building.

2. Secondary-Sodium and Steam Generator Building

This building houses the secondary-sodium pumps, secondary purification and drain systems, and the steam generators. Above grade the building is an enclosed steel structure extending approximately 120-ft high. External walls are corrugated aluminum panels. The building extends approximately 60 ft below grade.

The building is divided internally into four bays, three of which are approximately 75-by-45-ft, each containing a single secondary loop and its associated steam generators, and a fourth bay approximately 75-by-25-ft used for access. The below-grade portion of the building containing the steam generator relief equipment and lower end of the evaporator modules is partitioned by concrete walls into three separate enclosures. Above grade the bays are internally separated by fire-resistant paneling. This arrangement provides isolation in the event of a sodium leak or other malfunction and limits the consequence of such an incident to a single steam generator and secondary loop. Platforms are provided as required at various operating levels.

The building is located between the reactor building and the turbine building with the secondary-sodium system on the reactor side and the steam generating system on the turbine side. The combined secondary-sodium pump-case-expansion tank in each loop is located at the uppermost working level in the building so that the sodium level in the tank will be the high point in the secondary-sodium loop.

An elevator is provided for access. Cranes or other suitable handling equipment are provided for component maintenance and replacement.

3. Turbine-Generator Building

The turbine-generator building is adjacent to the steam generator structure. The building encloses the turbine-generator and main condensers, the boiler feed pumps, feedwater heaters, and associated turbine plant equipment.

The longitudinal axis of the turbine and the building runs parallel to a centerline extending through the reactor and steam generator structures. This arrangement protects the nuclear island from damage by missiles if blades are thrown by the turbine.

The building is a steel frame structure with enclosing walls of corrugated aluminum panels. The turbine-generator is set on a reinforced concrete structure with the turbine operating deck approximately 30 ft above grade. The main condensers are mounted within the concrete structure, directly below the turbine-generator. The main circulating water lines are below grade.

4. Fuel-Handling and Radioactive Waste Building

This is a multipurpose building located adjacent to the reactor building and connected to it by an equipment airlock. Facilities in this building provide for receiving and storing new fuel, transfer of spent fuel from the reactor building to a truck or railroad car for shipment off-site, the decontamination and packaging of small contaminated components removed from the reactor building, chemical analyses of test samples, and liquid and gaseous radioactive waste systems.

The building provides space for a 1/3-core loading of new fuel. The design is based on the fact that new fuel will be radioactive and direct contact with the fuel by personnel will not be permitted.

Equipment for handling and storage of radioactive liquid and gaseous wastes is located in shielded vaults below grade.

The building is constructed of concrete to provide shielding for the new fuel and other radioactive components which may be within the building. Access to the building will be by the airlock from the reactor building or from the outside by exterior doors.

5. Control Building

This building houses the plant control facilities consisting of the control room, cable spreading, and switchgear rooms beneath it. Instrument checkout and repair facilities are also included in this building. The building is of concrete construction to provide fire protection and radiation shielding. It is so designed that occupancy is possible during all postulated emergency conditions.

6. Heating and Ventilating Building

This building houses the reactor building heating and ventilating equipment and the exhaust air handling units and filters. The building is a steel frame structure with insulated metal wall siding.

7. Administration Building

This building contains administrative office spaces, training areas, and the plant personnel control area. The personnel control area contains a counting room, health physics offices, hot and cold change rooms, and activity

monitoring equipment. The major function of this area is to control and monitor all traffic entering and leaving potentially radioactive areas in the plant. The building is a steel frame structure with insulated metal wall siding.

8. Intake Structure

A concrete river-water intake structure is connected by an intake channel to the river or other source of circulating water. This structure houses the main condenser circulating water pumps, fire-extinguishment-water pumps, screen wash pumps, handling equipment, and other associated hardware.

9. Service Building

The service building is located at grade in close proximity to the turbine-generator building. This building houses a machine shop, pipe shop, sheet metal shop, etc., and serves as a central area for material receiving and plant maintenance, repair, and minor modifications.

10. Condensate Polishing Building

This building contains turbine feedwater makeup and treatment equipment. It is located at-grade next to the turbine building.

11. Boiler Building

The boiler building houses the plant auxiliary steam boiler and associated equipment.

12. Sewage Treatment

Sewage treatment facilities are provided for primary and secondary treatment of sanitary sewage.

13. Inert-Gas Area

The inert-gas area provides space for inert-gas storage tanks and associated equipment.

14. Circulating Water Discharge

A below-grade concrete structure and tunnel serves as the circulating water discharge to a canal and weir leading to the river at a point downstream from the intake structure.

B. REACTOR

1. General Description

The reactor is the heat source for the plant, generating 2500 Mw of thermal energy at nominal rating. It consists of a core of mixed plutonium and uranium oxide fuel, suitably contained and subdivided to assure adequate heat removal by the circulating liquid-sodium coolant. The core operates in a fast-neutron flux, and power is regulated by moving neutron-absorbing control elements into or out of the core. This arrangement is shown in Figure IV-4; the entire system is housed in a stainless-steel reactor vessel which provides containment for the sodium coolant, as shown in Figure IV-5. The open upper end of the vessel is closed by a thick double-rotating plug shield which is also part of the fuel-handling system. This shield is shown in Figure IV-6. At 100% load, sodium enters the vessel at the lower end at 760°F, flows up through the core, and exits from the vessel at 1060°F.

Storage for spent fuel is provided within the reactor vessel. An eccentric storage arrangement was selected to minimize the diameters of the two rotating shield plugs.

A concentric storage arrangement would have required the use of a three-rotating-plug shield to maintain the same large-plug diameter. Spent fuel is transferred from the reactor core to in-vessel storage positions without removing the fuel from the sodium. This feature, by simplifying fuel handling in the core, reduces to a minimum the reactor downtime required for this operation. Removal of spent fuel from the reactor storage into the ex-core storage area is accomplished in a separate concurrent operation.

Secondary containment of primary sodium is provided by the vessel containment jacket and cavity liners. The vessel jacket allows a void volume which is less than that of the volume of sodium in the reactor vessel above the outlet nozzles, thus precluding uncovering the core should there be a sodium leak from the reactor vessel. The cavity liner is 2-in. -thick to protect against missile penetration resulting from the DBA. The containment dome connected to the cavity liner provides the primary containment barrier fission-gas release from the reactor vessel following DBA (core meltdown.)

Reactor performance characteristics are given in Table IV-1.

TABLE IV-1
REACTOR PERFORMANCE DATA

Thermal-Hydraulic Performance			
Total reactor thermal power, Mwt (nominal)		2500	
Core power fraction (midcycle)		0.899	
Radial blanket power fraction		0.069	
Axial blanket power fraction		0.032	
Total reactor flowrate, 10 ⁶ lb/hr		93	
Peak nominal linear rod power, kw/ft (100% power)		15	
Radial peaking factor		1.26	
Axial peaking factor		1.25	
Reactor-mix mean-temperature rise, °F		300	
Reactor outlet coolant temperature, °F		1060	
Maximum nominal clad inside temperature, °F		1130	
Hot-spot clad inside temperature, °F		1200	
Average velocity highest-powered element, ft/sec		27	
Maximum velocity (at spacer), ft/sec		32.8	
Core pressure drop, psi		100	
Midcycle Equilibrium Nuclear Characteristics			
Number of fuel elements, fissile Pu wt %:			
Inner zone		118	(11.43)
Outer zone		138	(15.44)
Number of radial blanket elements:			
Inner zone		60	
Outer zone		66	
Core height, in.		50	
Axial blanket height, in.		12	
Radial blanket height (inner/outer, in.)		62/50	
Number of control rods:			
Shim-safety		9	
Safety		6	
Fissile masses (at midcycle equilibrium), kg:			
	Total	Pu	U ²³⁵
Core	1780	1758	22
Radial blanket	368	331	37
Axial Blanket	116	98	18
Total	2264	2187	77
Breeding ratio:			
Core		0.827	
Radial blanket		0.280	
Axial blanket		0.206	
Total		1.313	
Initial loading (Pu + U), kg:			
Core		13,339	
Radial Blanket		15,365	
Axial blanket		7,463	
Reactivity Coefficients			
Doppler constant, T dk/dt		-0.0096	
Maximum radial sodium void, %		6.82	
Isothermal sodium temperature coefficient, Δk/°F		3.3 x 10 ⁻⁶	
Prompt (Doppler) power coefficient [(10 ⁵ Δk/k)/% power]		-5.10	
Delayed (sodium) power coefficient [(10 ⁵ Δk/k)/% power]		+0.75	
Full-flow power coefficient [(10 ⁵ Δk/k)/% power]		-4.4

Figure IV-4. Reactor Structure
Cross-Section
(Dwg. N09620020)

Figure IV-5. Reactor Structure
Sectional Elevation
(Dwg. N09620018)

Figure IV-6. Reactor Structure
Plan
(Dwg. N09620019)

2. Reactor Core

The reactor core is made up of 256 vertical hexagonal elements of a 5.389-in. triangular lattice (see Fig. IV-4). The fast-fuel elements have a 50-in. active fuel section and 12-in. axial blanket sections at each end. The core is surrounded by two rows of radial blanket elements and a single row of stainless-steel reflector elements.

Fifteen control and safety rods are provided for reactivity control. The lower control-rod guide tubes have the same shape and dimensions as the fuel element housing.

Storage for spent fuel within the reactor is provided to handle slightly more than one-third of the fast-fuel elements. Radial blanket elements do not require in-core storage. The storage region is separated from the reactor core by 9 in. of stainless-steel thermal shielding.

3. Fast-Fuel and Blanket Elements

A fast-fuel element consists of a bundle of 217 fuel pins, vertically supported at their lower ends on support bars inside a hexagonal stainless-steel housing. The fuel pins are spaced laterally by eggcrate-type spacers at 4-in. intervals along their entire length. Fuel pellets of mixed PuO_2 and UO_2 , 0.215-in. -diam, are contained in 0.250-in. -OD 15-mil-thick stainless-steel cladding. The cladding is a gastight container for the fuel, and sufficient excess length is provided for the storage of fission gas to achieve the required fuel burnup. The inner and outer radial blanket elements are identical in configuration to the fast elements except that the inner blanket elements have 91 0.47-in. -diam pins and the outer have 61 0.58-in. -diam pins. The reflector elements are made of stainless-steel plates.

The fuel elements (see Figure IV-7) are positioned and supported at their lower end in a grid plate support structure. The tubular lower end or "nozzle" of the fuel elements fits into a tube in the support structure. Sodium flows into openings near the top of the nozzle and upward through the fuel bundle. Hold-down against the uplift due to sodium flow is provided by downward pressure against the closed lower end of the nozzle. Fixed orifices in the fuel-element nozzle assure proper distribution of coolant through the reactor. Sodium goes

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Figure IV-7. FBR Fuel Element Design Concept

from the fuel element at the upper end into the large pool which forms the outlet plenum of the reactor.

A gap is maintained between adjacent fuel elements to provide for assembly of the elements into the core array and their removal therefrom, and to accommodate manufacturing tolerances in the dimensions of the fuel-element housings. A tight core is achieved by hardfaced overlay spacer pads at each corner of the fuel-element housings, which provide interelement bearing points. The entire core array is laterally supported at the outer edge by scalloped radial support plates, mounted on the cylindrical stainless-steel shielding, which acts as cantilever beam springs that apply inward radial compressive forces. The elevation of the spacer pads and scalloped support plates is optimized to minimize core reactivity changes due to thermal bowing of the fuel elements.

Fuel elements are transferred between core and storage by an in-core fuel-handling machine which grips the upper end of the fuel element. Thermocouples are provided at the outlet of each fuel element.

4. Control Element

Reactivity is controlled by vertically moving rods in and out of the core which contain a neutron absorber. Tantalum has been selected as the reference absorber material. Nine control rods are used as combined shim and safety rods, and six are used strictly as safeties; thus a total of 15 provides the necessary shutdown margin. All rods can be inserted from any position and at any point in time.

The control-rod absorber, drive, and drive extension are shown in Figure IV-8. The control-rod assembly consists of the lower guide-tube, the absorber assembly, the actuator, and the drive. The external shape and size of the lower guide-tube are identical to that of the fuel-element housing, and the tube fits into the gridplate structure in the same way as a fuel assembly. The lower end of the guide-tube nozzle acts as a dashpot for the ram at the bottom of the absorber assembly.

The absorber assembly consists of a pull rod, an absorber column, and a snubber ram. During normal shim rod operation, the absorber column is moved vertically in the guide tube by the actuator rod. The control rod is

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Figure IV-8. AI FBR Control-Safety Rod Schematic

released by deenergizing the latch magnet; this allows the absorber assembly (with actuator detached) to fall freely into the core. Snubbing of the released rod is accomplished as the ram at the bottom of the absorber assembly enters the dashpot at the lower end of the guide-tube. During fuel handling, the absorber assembly is disconnected from the actuator so that the actuators and drives can be raised sufficiently to permit rotation of the top shield.

5. Reactor Vessel

The reactor vessel houses the core and provides containment for the sodium coolant and storage for new and spent fuel. The vessel (see Figure IV-5) is made of Type 304 austenitic stainless-steel and is 22-ft in diameter. It has an ellipsoidal bottom head and its upper open end is closed by a double-rotating shield plug. The vessel is supported from the top by a heavy stayed ring flange.

The vessel has three large inlet and outlet penetrations for the primary coolant loops and three smaller ones for control-rod and fuel storage cooling from the low-pressure region below the high-pressure inlet plenum. The low-pressure coolant is thermally driven to cool the control rods and stored fuel. This mode of cooling, called thermosyphon, is shown schematically in Figure IV-9. Temperature control for this cooling is provided by adjustment of the coolant bypass valves, to obtain the desired outlet temperature from the control rod channels.

The vessel internals consist of the gridplate support structure which also serves as the high-pressure inlet plenum, thermal-neutron shielding, and fuel storage thimbles.

The gridplate support structure is approximately 13-ft in diam and 3-ft high. It consists of two perforated plates connected by through-tubes on a 5.373-in. triangular pitch to accommodate the core assemblies. A cylindrical shell encloses this structure to form the high-pressure inlet plenum. The structure incorporates staybolts and stiffening members to reduce deflection to an acceptably low level. The core subassembly nozzles are inserted into the through-tubes and seat on conical surfaces in the lower grid-plate.

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Figure IV-9. Reactor Flow Schematic

Shielding to assure flux attenuation consists of 9 in. of cylindrical stainless-steel plate, supported on the gridplate support structure. The function of this shielding is to minimize fissions in the stored fuel, permitting the spent fuel to decay, to protect the reactor vessel from neutron damage, and to reduce internal heat generation in the external (biological) shielding.

A total of 91 in-core storage spaces are provided. The fuel storage thimbles have an CD of approximately 7.5 in. and support the stored fuel at the same elevation as subassemblies in the core. The center-to-center thimble spacing is 9 in. Spent-fuel subassemblies are cooled by thermal-convection flow of sodium.

The cover gas is helium, maintained at a pressure slightly positive to the operating area to prevent in-leakage. The cover gas system has the capability of removing fission gases in case of fuel pin failures. The reactor cover gas will be monitored for fission products.

6. Top Shield

The top shield is the upper closure of the reactor vessel. It provides radiation shielding for the operating floor area, a seal for the reactor cover gas, a heat-flow barrier, and mechanical support for the control-rod drives. It is a double-rotating plug (see Figures 5 and 6) and acts as an integral part of the fuel-handling system.

The larger of the two rotating plugs is concentric with the reactor vessel and turns on a thrust-bearing approximately 23-ft in diameter. The smaller plug is large enough to accommodate all of the control-rod drives and allow removal of the gridplate or other vessel internals as necessary. It is mounted on a separate thrust-bearing and has its axis midway between the center and outer surfaces of the larger plug. The smaller plug has a penetration near its periphery for the in-core fuel-handling equipment. The large plug has one penetration for each row of storage positions. Gear drives are provided to rotate each plug so that the fuel-handling machine can be aligned over any core or storage position. Positive indexing is provided by an electromechanical system similar to that for large tape-driven machine tools.

The overall thickness of the top shield is approximately 11 ft. The lower 15-in. section consists of horizontal layers of stainless-steel plate, the lower plates of which are submerged in sodium pool during normal reactor operation. Above this stack of steel plate is a 6-in.-high cover-gas space. The skirts supporting the steel stack are perforated to provide full communication for the cover gas.

The two large plugs as well as all penetrations through the top shield are stepped to prevent radiation streaming. Shield plug penetrations are required for the control-rod drive actuators. Static seals are provided for each of these penetrations. The seals for the two large plugs are lip seals internally pressurized to provide positive sealing during reactor operation. Friction is reduced during rotation of the plugs for fuel handling by reducing the internal pressure of these seals.

7. Fuel-Handling System

Fuel-handling operations constitute a major part of the scheduled reactor shutdown time and therefore are influential in establishing the upper limit on

plant availability. Fuel-handling operations have two distinct phases: (1) insertion of new fuel into in-vessel storage, removal of spent fuel from in-vessel storage, and preparation of spent fuel for off-site shipment; and (2) handling fuel within the reactor vessel to unload and reload the core. These are termed "ex-core" and "in-core" fuel handling, respectively. The guiding principles in the selection of the fuel-handling system and components are minimum scheduled downtime and simplicity for high reliability and ease of contact maintenance.

The reactor will be refueled on a six-month cycle, with $1/3$ of the core, $1/6$ of the inner blanket, and $1/9$ of the outer blanket being replaced each cycle. A like number of blanket elements will also be rotated 180° to permit uniformity of the Pu buildup associated with the sharp flux gradient at the edge of the core.

The fuel-handling system operation is as follows:

Refueling is done during plant shutdown. Spent fuel from the core region is lifted vertically with the in-vessel fuel machine and raised into the sodium pool above the core. The double-rotating plugs are rotated and the spent fuel is placed in an empty canister in one of the in-vessel storage positions for decay. During the next refueling period the spent fuel is removed from the in-vessel storage positions to the ex-vessel storage tank by the ex-vessel machine. From this tank it is removed for shipment. Blanket fuel is removed from the reactor without a decay period, but still requires both steps with the in-core and ex-core machines.

New fuel is first stored in canisters in the ex-vessel storage tank. Before loading, this fuel and canister is placed into an in-vessel storage position by the ex-vessel fuel-handling machine. During refueling the new fuel is transferred from the canister to the core with the in-vessel machine, and a spent-fuel element is subsequently loaded into this canister before removal.

The fast-core elements have an average reactor residence time of 1.5 yr, the inner radial blanket 3, the outer radial blanket 4.5, and the shim control rods 8 yr. This means that, on the average, after each 6-mo cycle 85.5 fast elements, 10 inner blanket elements, 7.3 outer blanket elements, and 0.5 control rods, for a total of 103.3 core assemblies, are exchanged. In addition, 15.5 blanket elements are rotated 180° . Only the fast-fuel elements and control rods need to be stored in the vessel. Only 86 in-vessel storage positions are

therefore required; however, 91 positions have been provided to add flexibility. Provisions have also been made in the ex-core fuel-handling machine and storage system to permit direct removal of fast elements when experience provides confidence in such a procedure. This capability would eliminate the need for on-site decay storage of spent fuel beyond the time dictated by shipping and/or reprocessing limitations, thereby minimizing fuel cycle inventory costs. Elimination of the need for in-vessel storage would also facilitate longer refueling intervals (annual refueling at a burnup of 100 Mwd/kg) and resulting improved plant availability. The downtime per refueling cycle is estimated at about 10 days. This results in a total scheduled unavailability of 5.6% due to fuel handling.

The overall configuration of the in-vessel fuel-handling machine is shown schematically in Figure IV-10. The machine has two main sections. The lower passes through the reactor top shield and reaches down nearly to the top of the fuel elements. This section contains the grapple, the guide-tube, and their actuators; it is installed in the reactor top shield by the auxiliary handling machine and the in-vessel-machine shielded gate-valve. The upper part consists mainly of the grapple and guide-tube vertical drive equipment and the structural framework supporting them. This part is set in place after the lower part has been installed, at which time the drives are connected.

The grapple is made with two fingers actuated by a rod with pins in its lower end operating in cam slots in the grapple fingers. Downward motion of the actuator rod moves the fingers outward, in which position they will engage the internal lip of a fuel element or control-rod housing. The actuator rod is sealed to the grapple tube by means of a bellows at the lower end and an elastomeric seal at the top end. Relative motion between the rod and the tube is caused by an air cylinder mounted at the top end of the tube. A guide-tube serves the dual purpose of guiding the lower end of a fuel element into an empty core or storage position and of holding down surrounding elements as a fuel element is removed from the core. When the guide-tube is fully down, it bears on the tops of the adjacent fuel elements and control-rod housings; when fully up it clears the tops of the fuel elements by about 4 in. The grapple rod is driven vertically by means of roller chains passing over upper and lower sprockets. The lower sprockets are driven through a gearbox by a variable speed dc motor;

Figure IV-10. In-Vessel Fuel-Handling Machine Schematic

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the upper sprockets are idlers. The grapple tube moves relative to the guide-tube, and the latter is long enough that no portion of the grapple rod wetted by sodium passes through the seals at the upper end of the guide-tube.

Both the ex-vessel fuel-handling machine and the auxiliary handling machine are mounted side-by-side on a common gantry. An operator's platform is supported from the trolley of the gantry. The gantry and ex-vessel fuel-handling machine are shown schematically in Figure IV-11. A hoist mechanism located at the top of the ex-vessel fuel-handling machine consists of three tape drums; two are load-carrying tapes and one is the grapple-actuating tape. The drums are located inside a shielded housing, one side of which can be removed for access to the drums, tapes, etc. The drums are mounted on shafts which pass through and are sealed to the shielding housing.

The ex-vessel machine grapple is basically similar to that of the in-vessel machine in the way in which the gripping fingers are actuated. The principal difference is that this grapple is made to grip only on the internal lip of the fuel element canister and will not grip a fuel element or a control-rod housing. This is a safety feature to prevent the accidental picking up of a spent-fuel element, which could overheat if removed from the sodium-filled canister.

When the ex-vessel machine is positioned over a floor-mounted shielded gate-valve, the entire machine is lowered a few inches until the seal is made. The lowering and subsequent raising are done by mechanical jacks on the gantry. The actual seal is made by means of a spring-loaded sliding sleeve within the valve body, and sponge-rubber-backed lead blocks insure that the shielding crack at the interface is minimized. This arrangement of sealing and shielding members makes it unnecessary for the full weight of the ex-vessel machine to be lowered onto the floor valve to make a suitable seal.

All core components are transferred into and out of the reactor vessel in canisters. Spent-fuel elements are also shipped out of the plant in canisters in which they were removed from the reactor vessel. Each canister is made of tubing 7-1/2 in. in diameter, closed at the bottom end and having a heavy ring welded on the top end. The ring section is machined with an internal pickup lip for the grapple of the ex-vessel. The ring section is also equipped with internal passages which will syphon out the top few inches of Na upon removal from a

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Figure IV-11. Ex-Vessel Fuel-Handling Machine Schematic

pool to reduce the likelihood of sloshing out of the canister during handling operations. When the canister is to be used for shipping spent fuel elements or other components a cap can be welded to the heavy ring section. The cap has identical pickup provisions so that the sealed canister can also be handled by the ex-vessel fuel-handling machine.

The grapple is sufficiently long to reach through the Na pool to the top of the fuel elements and control-rod housings without immersing the steel tapes in the liquid Na. This will greatly reduce the amount of Na that is carried up into the hoist housing. The grapple load tapes are of a high-strength steel, 4-in. - wide by 0.048-in. - thick. The grapple actuator tape is 3/4-in. - wide and of the same thickness. Before the tapes are wound onto the hoisting drums during operation they pass through scrapers that will remove most of any Na that may have condensed on them.

The cooling system for the ex-vessel fuel-handling machine uses a separate NaK-cooled inner sleeve within the main body of the machine. The latter is made with a double inner wall so that room air can be circulated on the outside of the inner wall and through the lead shot which comprises the shielding.

A closure valve at the lower end of the ex-vessel machine is essentially a gate-valve in which the gate is heavily shielded with lead and the body of the valve is filled with lead shot. The gate travels horizontally on rollers, driven by a lead screw and a standard commercial valve actuator. As the gate of the valve is driven to the closed position, a sealing plate is forced downward by a linkage arrangement to seal against the inside of the valve body. A drip pan, mounted atop the gate to catch any Na which drips from a canister in the machine or from the cooled inner sleeve, can be emptied by means of a drain valve. A heating coil built into the drip pan facilitates drainage of Na, which has solidified.

C. SODIUM HEAT-TRANSFER SYSTEM

1. General Description

The sodium heat-transfer system, shown schematically in Figure IV-12, consists of three identical heat-transfer circuits operating in parallel. Each circuit consists of a radioactive primary-sodium loop in series with a non-radioactive secondary-sodium loop. The primary-loop sodium flows through the reactor core, absorbing the energy generated, and transports the energy to the intermediate heat exchangers. The energy is transferred in the intermediate heat exchangers to the sodium in the secondary loop. This energy is then transported to the steam generator banks where it is transferred to the steam system. The heat-transfer system characteristics are given in Table IV-2.

Three equal $1/3$ -capacity heat-transfer circuits provide increased availability in the event of a component failure. The reactor can be operated at reduced power when only two of the three circuits are on-line. Should one of these two circuits fail, the remaining operable circuit is sufficient to provide for removal of decay heat by natural convection, and would protect the reactor following such an accident. Operation on only one loop is not permitted. Sodium flow for decay-heat removal is insured by the emergency power system and the backup natural-convection circulation through one or more of the circuits. The reactor, intermediate heat exchangers, steam generator complex, and interconnecting piping are arranged in elevation as shown in Figure IV-13 to provide the natural-convection flowpath.

All primary-loop components are located within shielded equipment cells in the reactor building. With the exception of the piping to the intermediate heat exchanger, the secondary-system components are located outside the reactor building structure between the reactor building and the steam generator area.

The operating pressure on the secondary-sodium (shell) side of the intermediate heat exchangers is maintained at a higher pressure than the primary-sodium (tube) side, so that a leak in the intermediate heat exchangers will not lead to contamination of the unshielded secondary system with radioactive sodium.

TABLE IV-2
HEAT-TRANSFER SYSTEM

Number of loops	3 primary and 3 secondary
Reactor thermal power, Mwt	2500
Total reactor flow, lb/hr	94×10^6
Reactor inlet temperature, °F	760
Reactor outlet temperature, °F	1060
Pressure at top of reactor vessel, psig	1
Primary-system sodium volume, ft ³	30,000
Primary-sodium volume/loop, ft ³	6,000
Secondary-sodium volume/loop, ft ³	14,000
Primary-sodium pumps:	3 (one/loop)
Flow, gpm	77,000
Head, ft	355
Brake horsepower, hp	6660
NPSH available, ft	30
Intermediate heat exchangers:	6 (two/loop)
Duty/IHX, Btu/hr	$1,425 \times 10^6$
Surface area/IHX, ft ²	16,600
Secondary-sodium pumps:	3 (one/loop)
Flow, gpm	87,200
Head, ft	250
Brake horsepower, hp	5,400
Steam generator banks:	3 (one/loop)
Evaporator duty/bank, Btu/hr	$1,820 \times 10^6$
Evaporator surface area/bank, ft ²	29,700
Superheater duty/bank, Btu/hr	510×10^6
Superheater surface area/bank, ft ²	12,800
Reheater duty/bank, Btu/hr	477×10^6
Reheater surface area/bank, ft ²	15,000
Sodium inlet temperature, °F	950
Sodium outlet temperature, °F	690

2. Primary-Sodium Loop

The combined primary-loop sodium flow is from the reactor lower plenum up through the reactor core into the upper plenum. The flow then divides and goes through three separate piping loops to a primary pump, through two intermediate heat exchangers in parallel and then back to the reactor lower plenum where it recombines with the sodium flow from the other loops. A small bypass flow is provided to the low-pressure region to cool the control rods and elements in storage positions. Flow through the loops is regulated to maintain nearly constant system temperatures throughout the power range. As the reactor power varies, the sodium flow in the loop is changed to avoid large temperature changes which could cause thermal shock and stress problems. At full-power operation 94×10^6 lb/hr of sodium enters the reactor at 760°F and leaves at 1060 .

Flow in each loop is provided by a vertical free-surface single-stage centrifugal pump similar to the Fermi and HNPF pumps, which utilize hydrostatic bearings lubricated by sodium supplied by the pump. The upper radial and thrust bearings are conventional oil-lubricated bearings. An oil-lubricated

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Figure IV-13. Reactor and Primary Loop Elevations

double mechanical seal is used at the upper cold end of the pump for atmospheric separation. Each pump is designed to deliver 77,000 gpm of 1060°F sodium at a 355-ft total dynamic head.

The pump driver is an adaptation of a modified Kramer system and is a combination 7500-hp wound-rotor induction motor and a 2500-hp dc motor, both directly coupled to the pump. The secondary of the wound-rotor motor feeds a silicon-controlled rectifier which in turn supplies power to the dc motor. The dimensions of the pump and driver are about 10-ft diam and 50-ft high. The exchange of power between the induction and dc motors provides for speed control of the drive to meet the controlled-flow requirements of variable power operation. Additional flow control can be obtained by a throttle valve located in each loop. The valve is used following a reactor trip to control flow and prevent overcooling of the reactor with subsequent loss of temperature gradients. These valves also provide a means of controlling flow at the low flows when the pumps are operating at minimum speed, or when convective flow is established. A check valve is installed in each loop to prevent flow reversal in the loop in case of loss of the loop pump or severe mismatch of the speeds of the three primary pumps.

Each primary loop has two vertical shell-and-tube intermediate heat exchangers operating in parallel. Each unit has a design duty of 1425×10^6 Btu/hr and is about 70-in. diam and 30-ft long. Primary sodium flows in the tubes.

Sodium degasification is accomplished in a small side loop of the heat-transfer system. Sodium is withdrawn from the reactor outlet plenum and the running vents from each of the six IHX's. Since it may be high in dissolved helium it is mixed with cold-leg Na to cool it and thereby reduce the He solubility. He which exceeds the solubility limit will evolve and be vented. The degassed Na is returned to the reactor inlet by one of two full-size degasifier return pumps.

The primary equipment and piping are of Type 304 stainless steel. The piping from the reactor vessel to the pumps is 48-in. diam and the return piping is 42. Pipe routing to and from the reactor and associated shielding is designed to minimize neutron streaming into the equipment spaces, thus

minimizing activation of equipment and secondary sodium in these spaces. Block valves are installed in the reactor inlet and outlet lines to permit loop isolation.

3. Secondary Heat-Transfer Loop

Each primary loop is in series with an independent secondary circulation loop. The flowpath in the secondary loop is up through the shell side of two parallel IHX's to the secondary surge tank which houses the secondary pump, down through a steam generator bank and back to the IHX's.

The secondary sodium in each loop is circulated by a vertical free-surface pump similar to the primary pump. The secondary pump is designed to deliver 87,200 gpm of 950°F sodium at a 250-ft total dynamic head. The pump requires 5,400 brake-hp to drive the pump at about 850 rpm. The pump driver is a combination 6,000-hp wound-rotor induction motor and a 2,000-hp dc motor. The secondary-sodium flow is controlled by pump speed in the same manner as the primary loop.

Block valves are provided to isolate the secondary-sodium equipment from the steam generators. The valve in the cold leg is a combination block-throttle valve which can be used to control convection flow following a scram.

The secondary surge tank is a 12,000-gal. vertical tank designed to hold the sodium volume change in going from isothermal at 600°F to 100% power operation. The secondary pump is located in the tank, drawing suction therefrom.

The steam generator bank in each loop consists of an evaporator section and a superheater section in parallel with a reheater section as shown in Figure IV-14. The steam generator design and operating characteristics are listed in Table IV-3. A throttle valve is located in the reheater outlet line to regulate the sodium-flow split between the reheater and the superheater. The three sections use the modular concept whereby a sufficient number of small units are connected in parallel to provide the required heat-transfer area. Each module is a vertical shell-and-tube unit with secondary sodium flowing in the shells.

7-S20-232-13A

Figure IV-14. Steam Generator Arrangement Schematic

TABLE IV-3
STEAM GENERATOR

Superheater module:			
Type construction		Shell-and-tube unit	
Total number in plant		3 banks	
Number of modules/bank		20	
Duty/module, Btu/hr		25.5×10^6	
Surface area/module, ft ²		640	
Tube data/module		79 tubes, 5/8-in. OD by 0.109-in. wall, 50-ft, 6-in. length	
		<u>Tube Side</u>	<u>Shell Side</u>
Fluid		Steam	Secondary Na
Flow/bank, lb/hr		2,500,000	18,400,000
Temperature in, °F		715	950
Temperature out, °F		905	860
Outlet steam pressure, psia		2515	-
Design pressure, psi		2700	200
Design temperature, °F		965	965
Material		2-1/4 Cr - 1 Mo	2-1/4 Cr - 1 Mo
Evaporator module:			
Type construction		Shell-and-tube unit	
Total number in plant		3 banks	
Number of modules/bank		39	
Duty/module, Btu/hr		46.8×10^6	
Surface area/module, ft ²		762	
Tube data/module		91 tubes, 5/8-in. OD by 0.095-in. wall, 51-ft length	
		<u>Tube Side</u>	<u>Shell Side</u>
Fluid		Water/Steam	Secondary Na
Flow/bank, lb/hr		2,500,000	35,500,000
Temperature in, °F		480	860
Temperature out, °F		715	690
Inlet feedwater pressure, psia		2600	-
Design pressure, psi		2800	200
Design temperature, °F		875	875
Material		2-1/4 Cr - 1 Mo	2-1/4 Cr - 1 Mo
Reheater module:			
Type construction		Shell-and-tube unit	
Total number in plant		3 banks	
Number of modules/bank		7	
Duty/module, Btu/hr		68×10^6	
Surface area/module, ft ²		2140	
Tube data/module		161 tubes, 1-in. OD by 0.095-in. wall, 50-ft length	
		<u>Tube Side</u>	<u>Shell Side</u>
Fluid		Steam	Secondary Na
Flow/bank, lb/hr		2,340,000	17,100,000
Temperature in, °F		553	950
Temperature out, °F		905	860
Outlet pressure (steam), psia		505	-
Design pressure, psi		600	200
Design temperature, °F		965	965
Material		2-1/4 Cr - 1 Mo	2-1/4 Cr - 1 Mo

The secondary surge tanks and pumps, intermediate heat exchangers, and piping up to the steam generator complex are constructed of Type 304 SS. The steam generator sections with interconnecting piping are constructed of 2-1/4 Cr - 1 Mo steel.

4. Steam Generator Bank

The design provides one bank of evaporator and superheater modules for each of the three secondary loops. Each bank is arranged separately in an in-line array, with 20 superheater and 39 evaporator modules. They are mounted vertically to minimize space requirements and avoid sodium stratification during free convection. Pressure-relief headers are provided to handle sodium-water reaction products should a leak occur; collection headers from the modules discharge to pressure-relief tanks. The evaporator and superheater modules are supported on a steel structure.

Each evaporator module consists of a 10-in. Schedule-20 pipe shell containing a tube bundle with 91 tubes, each 5/8-in. -OD and 0.095-in. -wall. Shell-and-tube material is 2-1/4 Cr - 1 Mo. Each superheater module consists of a 10-in. Schedule-20 pipe shell containing a tube bundle with 79 tubes. These tubes are 5/8-in. -OD and 0.109-in. -wall. The tube-and-shell material is 2-1/4 Cr - 1 Mo. Each evaporator module is designed to produce 64,000 lb/hr of steam at 715°F and 2600 psi. Each superheater module is designed to raise 125,000 lb/hr of steam from the evaporator to 905°F and 2500 psi.

A bank of reheater modules is located next to the evaporator and superheater modules in the same support structure. Similar to the other modules in construction, seven reheaters are in each of the three secondary loops. Each reheater module consists of an 18-in. -OD shell with 1/4 in. -wall, and a tube bundle of 161 tubes, each 1-in. -OD by 0.095-in. -wall. The shell-and-tube material is 2-1/4 Cr - 1 Mo. Each reheater module is designed to heat 337,000 lb/hr of steam from 540 to 905°F.

Each module has a hydrogen detector mounted on the sodium outlet piping to detect possible water leakage into the sodium. It is believed that a H₂ detector capability for sensing as little as 3 ppb of H₂ in Na with a response time of a few seconds can be developed. Normally a leak will be detected at the early stages of failure and the affected bank shut down before the leak becomes

significant. However, rupture discs are provided to relieve excess pressure in a tube failure and to vent the sodium/steam/gas mixture to the pressure-relief system. The relief system separates the liquid Na from vapors and holds it in a dump tank. The vapors are discharged to the stack. The steam generator isolation and recovery system is discussed in more detail later in Section F.

D. TURBINE-GENERATOR PLANT

1. Basic Design

The reference plant net electrical output at the generator terminals is a nominal 1000 Mw. The turbine-generator is a cross-compound unit operating in a reheat regenerative cycle with seven feedwater heaters, as shown in the heat balance diagram, Figure IV-15. The main turbine is a 3600/1800-rpm cross-compound unit with 43-in. last-stage buckets, designed for initial steam conditions of 2400 psig and 900°F, with reheat to 900°F at 500 psig. The rated capability of the turbine at 1.5-in. Hg abs condenser pressure and 0% makeup is 1066 Mw under normal operating conditions. The generator capability at rated hydrogen pressure and 0.85 power factor is 1240 Mva. Steam from seven extraction points is used for heating the feedwater to a final temperature of 478°F at 100% turbine flow.

Steam from the intermediate pressure turbine exhaust is supplied at 190 psig and 675°F to two turbines which drive two half-capacity boiler feed pumps. Each turbine exhausts to its respective condenser operating at 2-in. Hg abs. High-pressure steam at 2400 psig 900°F is used for final stages of startup, low-load operation, and overload conditions. Startup may also be initiated using steam from two auxiliary fossil-fuel-fired steam generators.

The normal path of condensate and feedwater flow from the single-shell condenser to the steam generators is as follows: condenser hotwell, condensate pumps, condensate coolers, condensate prefilters, condensate mixed bed demineralizers, generator hydrogen coolers, generator stator coolers, turbine gland steam condensers, low-pressure feedwater heaters, deaerating feedwater heaters, boiler feed pumps, and high-pressure feedwater heaters. The plant power generation, thermal requirements, and efficiency are given in Table IV-4.

2. Turbine-Generator

The unit is rated 1240 Mva at the generator terminals, with inlet steam conditions of 2400 psig, 900°F, reheat to 900°F, and exhaust to 1.5-in. Hg abs condenser pressure and 0% makeup. It is of the cross-compound type with the high-pressure shaft at 3600 rpm and the low-pressure at 1800.

TABLE IV-4
TURBINE PLANT DESIGN SUMMARY,
100% STEAM FLOW

Gross generator output (0.85 pf), Mwe)	1052
Plant auxiliary power, Mwe	47.8
Plant net electrical output, Nwe	1004
Plant net heatrate, Btu/kw-hr	8375
Plant net thermal efficiency, %	40.8
Evaporators thermal power, Mwt	1601
Superheaters thermal power, Mwt	449
Reheaters thermal power, Mwt	419
Turbine steam conditions, psia/°F/°F	2415/900/900
Steam flow, 10 ⁶ lb/hr	7.536
Condenser pressure, in. Hg abs	1.5

The 3600-rpm turbine consists of a double-flow high-pressure element and a double-flow intermediate-pressure (reheat) section. Steam is admitted to the high-pressure section through four stop and four control valves. Exhaust steam from this section at 575 psia, 555°F is reheated in the sodium-to-steam reheater and returned to the intermediate-pressure element through four combination stop-intercept valves at 515 psia 900°F. A portion of this exhaust steam is extracted for feedwater heating in the No. 7 heater. Extraction steam for the No. 6 is taken from the intermediate pressure casing. Exhaust steam from this turbine is used for feedwater heating in the No. 5 heater, and for operating the two boiler feed-pump turbines. The 54-in. -diam crossover pipes carry the rest of the exhaust steam to the two 1800-rpm low-pressure sections.

The 1800-rpm low-pressure turbine consists of two identical double-flow sections which exhaust downward through their respective connections into the single-shell condenser. Extraction steam for the deaerator and the No. 1, 2, and 3 feedwater heaters is taken from the low-pressure casings. The last-stage buckets are 43 in. long. The exhaust moisture does not exceed 10% under the rated normal operating conditions. Each low-pressure exhaust casing is provided with water-spray nozzles and control valves to prevent exceeding the

Figure IV-15. Heat Balance Diagram
(Dwg. No 9630009)

maximum recommended exhaust hood temperature during startups, shutdowns, and abnormal operating conditions.

The 3600-rpm generator is rated 620 Mva at 0.85 pf, 60-psig hydrogen pressure, 3-phase, 60 cps, and 22 kv. The 1800-rpm generator is rated the same except for hydrogen pressure, which is 45-psig. Both generators are designed for a short-circuit ratio of 0.58.

The generators are totally enclosed to operate with hydrogen gas as the principal cooling medium. This system, including fans and coolers, is contained within the generator. The stator windings, which have hollow copper conductors, are cooled by purified de-ionized water circulating within the coils. Heat exchangers outside the generators are used to reject the cooling system heat to the condensate water.

Dc power for excitation is supplied by a static exciter system. At any load point the load distribution between the two generators depends on the turbine characteristics. The power factor of the two generators is controlled by individual field excitation.

a. Control System

The turbine-generator has an electrohydraulic control system with electronic computing devices and a fire-resistant hydraulic system to actuate the steam valves. The control system consists essentially of the following functions:

- 1) Speed control;
- 2) Full-arc to partial-arc transfer;
- 3) Load control;
- 4) Initial pressure limit;
- 5) Stop-valve control flow;
- 6) Control-valve flow control;
- 7) Intercept-valve flow control;
- 8) Turbine bypass valve control; and
- 9) Overspeed limit.

b. Auxiliary Systems

The turbine-generator has the following auxiliary systems:

- 1) Bearing lubrication oil;
- 2) Control fluid;
- 3) High and low speed shaft turning gear;
- 4) Turbine steam seal and leakoff;
- 5) Turbine steam seal regulator;
- 6) Generator hydrogen cooling and sealing; and
- 7) Generator conductor cooling water treatment and control.

3. Boiler Feed-Pump Drives

Two steam turbines are used to drive the two half-capacity boiler feed pumps. The turbines are provided with dual inlets which admit low- or high-pressure steam according to the load requirements. At plant loads exceeding approximately 60% of rated, the turbines receive low-pressure superheated steam from the intermediate-pressure turbine exhaust. High-pressure steam at 2400 psig and 900°F is used below 40% of rated load, hot startups, and overload conditions. Dual steam admission occurs between 40 and 60% of rated load. Startup may also be obtained using steam from the auxiliary steam generators.

Each unit is provided with turning gear, control system, lubricating and control oil systems for both the turbine and the pump, and monitoring equipment. The sealing steam and drain systems are connected and controlled from the main turbine gland seal and exhaust system.

4. Main Condenser

The main condenser is a horizontal single-shell unit with a single circulating-water pass and divided epoxy-coated steel water boxes. The unit is solidly mounted on foundations with its axis parallel to the turbine-generator shaft. Expansion movement between the condenser and the low-pressure turbine casings is provided by belt-type expansion joints. The No. 1 low-pressure feedwater heaters are mounted in the two condenser steam hoods. The

deaerating-type condenser hotwell provides sufficient capacity to allow approximately one minute of operation at rated load.

The condenser design provides for receiving the condensate from the boiler-feed-pump turbines, the low-pressure heater drains, emergency drains, condensate pump recirculation, condensate expansion, and demineralized water supply. It also provides for receiving the steam from the superheater bypass, main steam turbine bypass, and reheat steam turbine bypass, up to 15% of rated flow for a limited time during plant startup and sudden load reductions.

A vacuum system is provided for evacuation of the water side of the condenser shell, to reduce the startup total dynamic head requirements of the circulating water pumps before establishing full-flow syphon. The design for the condenser and its auxiliaries is based on a fresh cooling-water supply, and the materials have been selected accordingly.

a. Circulating Water Pumps

Two motor-driven vertical mixed-flow pumps, which circulate 320,000 gpm of 55°F cooling water through the condenser, are designed to produce a total dynamic head of 25 ft at 270 rpm and 87% efficiency. The pump casings are of fabricated steel construction, the rotors have chrome steel shafts and bronze impellers. The motors are rated 1250 shp each at 270 rpm and 4160 volts. Weather-protected NEMA Type II enclosures are used.

b. Hotwell Pumps

Four motor-driven vertical centrifugal pumps, one a spare, circulate the condensate from the hotwell to the deaerating heater. Each pump delivers 3960 gpm of 92°F water at 800-ft total dynamic head with an efficiency of 85% at 1180 rpm. The pump casings are cast iron, the rotors have stainless-steel shafts and bronze impellers. The motors are rated 1000 shp each at 1180 rpm and 4160 volts. They have drip-proof enclosures and Class B insulation.

c. Air Removal

Air and noncondensable vapors are removed from the condenser by three positive displacement pumps, one a spare. Each pump has a holding capacity of 15 scfm at 1-in. Hg abs, and is driven by a 125-shp motor operating at 575 rpm and 460 volts. The pumps are cast iron with stainless-steel shafts.

5. Auxiliary Condensers

Two auxiliary condensers receive the exhaust from the two boiler feed-pump turbine drives. Each condenser is a horizontal unit, with double circulating water passes and divided epoxy-coated steel water boxes.

Each unit contains 2200 tubes with a total surface of 9500 ft²; the tubes are 7/8-in. -diam, 18 BWG, with an effective length of 20 ft. The condenser materials are the same as those used in the main unit. This design permits operation at a condenser pressure of 2-in. Hg abs with a flow of 10,800 gpm of 55°F circulating water, and at the conditions existing at the 100% main turbine flowpoint.

The air-removal and cooling-water systems are common with the main condenser. Condensate from the auxiliary condensers is pumped to the main condenser hotwell by the auxiliary condensate pumps.

a. Auxiliary Hotwell Pumps

Three motor-driven vertical centrifugal pumps, one a spare, circulate the condensate from the two hotwells to the main condenser hotwell. Each delivers 300 gpm of 101°F water at 50-ft total dynamic head with an efficiency of 70% at 1750 rpm. The pump casings are cast iron, the rotors have stainless-steel shafts and bronze impellers.

The motors are rated 7.5 shp each at 1180 rpm and 480 volts. They have drip-proof enclosures and Class B insulation.

6. Condensate and Feedwater Systems

The thermal cycle is based on seven stages of feedwater heating; three low-pressure and three high-pressure stages are located ahead of and after the deaerating heater, respectively. The condensate and feedwater stream is divided into two parallel trains of heaters.

Bypasses and interconnecting headers permit operation during startup and heater-outage conditions. The high-pressure heater drains are cascaded to the deaerator, the low-pressure to the main condenser.

Full-flow filtration and demineralization of the condensate are used to provide feedwater of the highest quality to the once-through steam generators.

Heater-tube leakage is minimized by using a fully welded design. Copper and zinc pickup problems are mitigated by the use of ferrous materials in the feedwater heating equipment.

Two half-capacity turbine driven pumps circulate the feedwater from the deaerating heaters to the three steam generators. The pumps are paralleled through suction and discharge headers to minimize unbalance between the heater trains. The turbine drives permit flexibility during startup and normal operation. Feedwater flow control to the steam generators may be accomplished over the normal load range by a combination of individual valve-control and pump-speed regulation.

a. Closed Feedwater Heaters

Both the six low-pressure and six high-pressure heaters are arranged in two parallel trains of three. All heaters except the deaerator are closed U-tube two-pass removable-shell units sized for 1/2-capacity each. The heaters have carbon steel shells and tubing, with 100% tube welding to the tube sheets. A nitrogen blanketing system prevents heater corrosion during outage periods.

The heaters are numbered in the direction of feedwater flow; the lowest-pressure heater, No. 1, is located in the condenser steam hoods. The total pressure drop through the three high-pressure heaters at 100% flow is approximately 50 psi; that through the three low-pressure heaters is approximately 65. The tube-side velocity at the design flow is approximately 8 fps. The heaters are designed with 10°F drain coolers.

b. Deaerating Feedwater Heaters

The two deaerating heaters consist of horizontal shells for heating and deaerating purposes mounted upon horizontal storage tanks. The storage tanks have a capacity to supply approximately 4 min. of feedwater at rated load. Four downcomer pipes link the heater and storage tanks. The deaerators are located on top of the steam generator building.

The deaerator heaters may be pegged from the No. 6 heater at low loads or from the flash tank during startups and shutdowns.

c. Boiler Feed Pumps

The two half-capacity turbine-driven pumps are barrel-type 5-stage units. Each has a capacity of 3.768×10^6 lb/hr (8070 gpm) at a total dynamic head of 8540 ft when operating at 5500 rpm. The NPSH is approximately 125 ft with 300°F water from the deaerater storage tanks. The pump efficiency at the preceding operating conditions is 84%.

The pump casings are of carbon steel with chrome steel inner casing, impeller, and shaft. Each pump is supplied with adequate lubricating oil from the steam-turbine oil system. An automatic recirculation control system is used to protect the pumps under startup, low-load, or no-flow conditions.

7. Water Treating Plant

The water treating facilities consist of two distinct demineralizing plants. One plant is for feedwater makeup to the main-turbine steam-generator cycle, and is designed for approximately 3% of the rated turbine throttle flow. The second plant performs the 100% condensate-flow polishing function.

The makeup system consists of three strings of single-bed cation exchangers followed by three strings of single-bed anion exchangers, with a final polish in three parallel mixed-bed units. Capacity of this makeup system can be obtained with the parallel operation of two of the three strings, while the third is in the regeneration cycle. Tanks for sulfuric acid, sodium hydroxide, and storage are provided.

The condensate polishing system consists of a parallel bank of prefilters followed by the mixed-bed demineralizers and a parallel bank of afterstrainers.

8. Main and Reheat Steam

The flow diagram of the main and reheat steam systems, shown in Figure IV-16, contain the piping which interconnects the steam generators, the main turbine, and the boiler feed-pump drives. Pipe sizes were based on steam velocities ranging from 170 to 220 ft/sec at the 100% steam flow conditions; thickness was determined according to the American Standards Association Code for Pressure Piping.

Figure IV-16. Flow Diagram,
Main Steam and Reheat Steam
(Dwg. N09630010)

Table IV-5 shows the design conditions and materials used in the main and reheat steam systems.

TABLE IV-5

MAIN AND REHEAT STEAM PIPING DESIGN DATA,
100% STEAM FLOW

Service	°F	psig	ASTM	Grade
Main steam	900	2575	A369	FP22
Hot reheat	900	675	A155	1 to 1-1/4 Cr
Cold reheat	600	675	A155	KC70
Low-pressure to boiler feed-pump turbines	675	250	A106	B
High-pressure to boiler-feed-pump turbines	900	2575	A335	P22

E. ELECTRICAL SYSTEM.

The electrical single-line design, shown in Figure IV-17, conforms to the best modern nuclear central station practice with the emphasis on reliability and continuity of service.

1. Turbine-Generator System

The turbine-generator is a cross-compound unit, each generator rated at 620 Mva, 0.85 pf, 22 kv, 60 cps. Each has a solid-state exciter. The output of the generators is connected to the station stepup transformers and to the station service auxiliary transformers through forced-air-cooled isolated-phase bus ducts.

2. Auxiliary Power Systems

Auxiliary voltages are 4160, 277/480, and 120/208 volts ac; 125 and 48 volts dc (user preference for 6.9 kv and/or 250 volts dc can be accommodated).

Sectionalized 4160-volt buses are fed from the normal station service transformer (connected by removable leads to the delta section of the main generator) or from the reserve station transformer (assumed 138/4.16 kv). Transfer will be automatic and of the fast "dead-bus" type.

The 480-volt buses are also sectionalized, with feeds from 4160/480-volt transformers. Each vital service control center is connected to one of the emergency 480-volt buses and one of the normal-service 480-volt buses.

Power is transformed as necessary to provide 120/208-volt service for receptacles, lights, and other low-voltage requirements.

3. Emergency Power Systems

Emergency 480-volt power is provided by four 750-kw gas-turbine generators. One generator is connected to each emergency and vital service 480-volt bus. Start is automatic on undervoltage (with time delay), with stabilized supply in less than 30 sec.

Two banks of station batteries are provided, each rated at 125 volts, 3000 amp hr, at the 8-hr rate. Each battery bank is charged by a static-type rectifier. The station batteries supply dc control circuits, emergency power

Figure IV-17. Electrical Single-Line Diagram
(Dwg. N09670003)

for various vital equipment drive motors, static inverters for the uninterruptable 120/208-volt buses, and static inverters for isolated instrument buses. A separate 48-volt dc supply is provided for computer and logic circuits (24-volt user preference can be accommodated).

The static inverters for uninterruptable 120/208-volt service are continuously paralleled with normal plant ac power, and run unloaded except for trickle charge unless normal ac power is not of suitable quality or is lost completely. Inverters for instrument service are directly connected to the battery. An installed common standby inverter can be switched to any instrument bus.

4. Lighting, Communications, and Annunciation

Lighting will be in accordance with the latest Illuminating Engineering Society recommendations as modified by user standards. Emergency lighting is supplied from the vital service buses.

The telephone and public address systems will be run in a separate conduit. A public address, paging, and communications system will be provided.

Considerable attention will be paid to the annunciator and central display system. It will include both "first-out" windows and events-recording as part of the plant computer.

5. Wire and Cable, Conduits and Trays

Wire and cable will be according to user standards, with insulation rated for 600 volts and 5 kv. Largest possible use of cable trays will be made, with proper attention to separation of power and instrument cables. Special attention will be given to isolation of nuclear instrument leads. Shielded 5-kv cables will be run underground in fibre duct encased in concrete. Conduit for lighting will be electrical-metallic tubing. Other conduit will be galvanized iron or aluminum.

6. Motors

Motors larger than 250 to 300 hp (as agreed with the user) will operate on 4160 volts. Those motors connected to the general service and emergency switchgear will use electrically operated circuit breakers for starters. All other motors will feed from combination circuit-breaker starter-motor control

centers. Motors will generally be NEMA Design B with Class A insulation and drip-proof enclosures. Motors installed outdoors will generally be weather-protected or totally enclosed, with heaters if required.

7. Grounding and Cathodic Protection

A station ground of less than 1-ohm is recommended. All piping and conduit, fences, building steel, motor frames, and other metallic structures will be connected to the ground system. A separate instrumentation ground grid will be provided and connected to the station ground at only one point. Insulating flanges and test connections will be installed on underground piping for tests at the site to determine cathodic protection required.

F. CONTROL AND INSTRUMENTATION

Control and instrumentation are described here in outline form only. Additional descriptive material will be given in the Final Task III report.

1. Plant Control System

The function of the plant control system is to maintain system temperature and flows within allowable ranges during normal operating and emergency shutdown conditions. Control of the plant will be from a central room located outside the reactor building. A cascaded load-following control system will be used. Normal operation will be with the turbine admission valve controlled by the speed governor in response to demands of area load frequency control.

2. Plant Protection System

The protection system is to prevent damage to the plant in case of maloperation, component failure, or malfunction, without compromising the continuity of plant service by unnecessary shutdown. This will be accomplished by redundancy in both sensors and components, and in level of protection.

3. Radiation-Monitoring System

This system is designed to protect plant personnel and the general public against damaging radiation. Additionally, criticality monitoring is supplied for fuel storage and handling areas. Additional instrumentation assists in determination of breeches of containment.

G. AUXILIARY SYSTEMS

Auxiliary systems are also described here only in outline form. Additional descriptive material will be provided in the Task II and Task III reports.

1. Startup and Steam-Dump System

The startup system is provided for placing the three banks of once-through steam generators and the turbine into operation after either a normal startup or a restart sequence following a turbine trip.

The steam-dump system is provided to release to the main condenser the excess energy stored in the reactor, steam generator, and reheater following a turbine-generator trip or sudden load reduction.

2. Sodium Preheating System

The function of the preheating system is to heat the sodium system and steam generator by external means.

All sodium systems are preheated to about 350°F to ensure that the sodium will remain molten during the initial filling operation. The steam generators are preheated by circulating treated water through the water-steam portion of the generators. The preheat system will also keep the sodium-containing systems hot as required for auxiliary operations and during extended shutdowns.

3. Sodium Storage and Treatment System

The basic functions of the sodium storage and treatment system are to:

- 1) Receive, melt, filter, and store reactor-grade sodium;
- 2) Fill and drain the main primary and secondary heat-transfer loops;
- 3) Purify the sodium in the primary and secondary heat-transfer loops;
- 4) Provide a source of sodium for filling the fuel-decay-storage tank; and
- 5) Monitor sodium impurities in the main heat-transfer loops and fuel storage tank.

4. Steam Generator Isolation and Recovery System

The steam generator isolation and recovery system accepts the products resulting from a steam or water reaction with sodium in a steam-generator module tube leak or rupture. The system also relieves any excessive pressure, thus preventing damage to other steam-generator modules or to the secondary-sodium system.

5. Post-DBA Emergency Cooling System

This system is an emergency cooling system for preventing loss of containment in a DBA core-meltdown accident. Due to afterglow heat generation in the fuel mass, the bottom of the reactor cavity must be cooled to prevent the core-meltdown products from melting through the cavity liner. During normal operation a small amount of heat generated by radiation will also be removed.

6. Helium Gas Supply System

This system provides a source of inert gas. Helium is used primarily as the cover gas for the primary- and secondary-sodium systems.

7. Nitrogen Gas Supply System

This system provides a source of oxygen-free gas. Nitrogen is used primarily as the inert atmosphere in the primary-sodium system cells and vaults.

8. Heating and Ventilating System

This system provides ventilation, filtering, monitoring, exhausting, temperature control, and humidity control for air in the buildings and various equipment cells.

9. Kerosene Cooling System

Primary function of this system is to service the space cooling system within the reactor building. Kerosene is used to minimize the quantity of water in the building.

10. Radioactive Vent System

The objective of the radioactive vent system is to handle gaseous radioactive fission products, activated gases, entrained radioactive particles, or a combination of any of these.

11. Radioactive Liquid Waste System

This system is provided to collect, store, and dispose of aqueous radioactive waste materials from the plant.