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AI-AEC-12792 (Vol. I)

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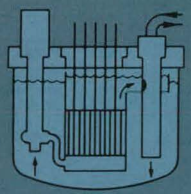
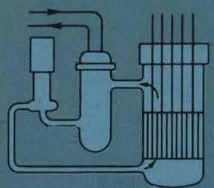


1000 MWe

L M F B R

Liquid Metal Fast Breeder Reactor



FOLLOW-ON STUDY

ATOMICS INTERNATIONAL

TASK III REPORT

Conceptual Design Report Summary

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Prepared for ARGONNE NATIONAL LABORATORY

United States Atomic Energy Commission Division of Technical Information

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AI-12792 (Vol. I)
REACTOR TECHNOLOGY (TID-4500)

1000-Mwe LIQUID METAL FAST BREEDER REACTOR
FOLLOW-ON STUDY
CONCEPTUAL DESIGN REPORT

VOLUME I
SUMMARY

ATOMICS INTERNATIONAL

A DIVISION OF NORTH AMERICAN ROCKWELL CORPORATION

CONTRACT: ANL 31-109-38-1996

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I. INTRODUCTION

Atomics International (AI) is one of the five contractors on the 1000-Mwe Liquid Metal Fast Breeder Reactor (LMFBR) Follow-On Study Program, which is being managed by Argonne National Laboratory (ANL) for the Atomic Energy Commission (AEC). This work is part of the AEC's continuing LMFBR design studies, as outlined in "Volume 2, Plant Design, LMFBR Program Plan" (WASH-1102). The overall objectives of this program are to identify the research and development (R&D) necessary to lead to safe, reliable, and competitive LMFBR central station power plants in the 1980's, and to establish the relative ranking of the needed R&D. The study objectives are being met through the development of a series of industry-favored reference plant designs and the evaluation of these designs, to determine the R&D program necessary to proceed with the final engineering and construction of the proposed plants. This report discusses the engineering studies which led to the AI Task III final reference design, and describes this design.

A. STUDY PROGRAM TASKS

The Follow-On Study Program was divided into four tasks by ANL. Task I comprised the concept selection and the establishment of all significant design criteria. Under this task, the major trade studies performed on the Company program were re-examined, in light of the criteria for the Follow-On Study Program. These trade studies included

1) Pool vs loop
2) Fuel handling
3) Vented vs nonvented fuel
4) Number of loops.

Because of their importance in setting the basic plant configuration and its performance characteristics, these tasks were reviewed in Task I. The selection of mixed-oxide fuel and a regular core geometry was based on previous work, and was not reevaluated. The next step was to define a preliminary reference design for a 1000-Mwe plant, based on this prior work, and with design and performance characteristics similar to those for the current AI Demonstration Plant reference design. The results of the Task I effort were reported in AI-AEC-12765, "ANL 1000-Mwe LMFBR Follow-On Study Task I Report."

Tasks II and III were done concurrently. The Task III work comprised a safety evaluation, core and system parametric studies, and alternate plant and equipment configuration studies. These studies have led to the Task III final reference design described in this report. A plant capital cost estimate, a fuel cycle cost analysis, and an overall economic assessment of the Task III final reference design was performed in Task III also. Task II was the preparation of Conceptual System Design Descriptions for the Task III final reference design. These are given in AI-AEC-12791, "1000-Mwe Liquid Metal Fast Breeder Reactor Follow-On Study Conceptual System Design Descriptions."

The Task IV report, AI-AEC-12793, "1000-Mwe Liquid Metal Fast Breeder Reactor Follow-On Study Research and Development Requirements," will describe the information desired before final design and construction of the proposed plant can begin, and will outline the necessary R&D which can most effectively provide the information.

B. STUDY APPROACH

The "LMFBR Program Plan — Volume 1, Overall Plan," (WASH-1101) provides important guidance on the relationship between the 1000-Mwe LMFBR Study Program and on-going individual demonstration plant programs, as follows:

> "Major reactor manufacturers, in conjunction with utility groups, are studying demonstration plants for near-term commitment. These manufacturers, therefore, are studying both near-term demonstration plants and longer-range target plants. Accordingly, the reference designs resulting from the AEC-funded studies of target plants are expected to be closely related to those from corresponding studies by industry of demonstration plants".

This guidance is further amplified in Volume 2 (WASH-1102), which states,

> "the designs of the demonstration plants should bear a close resemblance in many respects to the reference designs of the target plants."

Atomics International, in developing its 1000-Mwe reference designs, has closely coordinated the Follow-On Studies with the Company-sponsored demonstration plant program, as well as with the recently completed study of the technical and economic performance of a minimum R&D 1000-Mwe FBR, performed for the Empire State Atomic Development Associates (ESADA). The technology relationship of the AI Demonstration Plant to the Task I preliminary reference design and the Task III final reference design is illustrated by similarity of design features and performance characteristics shown in Table I-1. It is seen that the Task I design represents a modest extrapolation of technology from the Demonstration Plant, except for scaleup of equipment sizes, while the Task III design involves higher system temperatures, and thus anticipates successful completion of additional R&D.

C. PROGRAM LOGIC

When work was initiated on the Follow-On Study, most of the key trade studies had been completed as part of the Company-sponsored Fast Breeder Reactor Program. The conceptual design of the Demonstration Plant was in an advanced stage; and a parallel study of a 1000-Mwe LMFBR plant was in progress, under ESADA sponsorship.

The following were the principal guidelines used in selecting the Demonstration Plant design:

1) Maximum use of state-of-the-art and proven technology, in order to obtain a low-risk demonstration plant (this resulted in conservative design ratings, such as 1200°F cladding temperature limit, use of ferritic material for superheaters, etc.)

2) Schedule based on a 1974-1975 startup date, thus dictating completion of the required R&D program by about 1970

3) Conservative safety philosophy, which is believed to be required for the LMFBR demonstration plants

4) Applicability to future larger plants which are competitive with other power sources.

The primary objective of the ESADA study was the definition of the technical and economic characteristics of a 1000-Mwe plant design which would represent

TABLE I-1

COMPARISON OF PLANT CHARACTERISTICS

Parameter	500-Mwe Demonstration Plant	1000-Mwe Task I Design	1000-Mwe Task III Design
Reactor Rating (Mwt)	1250	2500	2400
Net Station Heat Rate (Btu/kwh)	8400	8375	8175
Fuel Material	Mixed Oxide	Mixed Oxide	Mixed Oxide
Reactor Outlet Temperature (°F)	1060	1060	1140
Maximum Nominal Linear Fuel Pin Power (kw/ft)	15	15	16
Maximum Nominal Cladding Temperature (°F)	1130	1130	1220
Core Height (in.)	50	50	43
Fuel Element	Hexagonal	Hexagonal	Hexagonal
Number of Pins	217	217	217
Pin OD (in.)	0.25	0.25	0.30
Distance Across Flats (in.)	5.323	5.339	5.617
Control Rod Absorber Material	Tantalum	Tantalum	Tantalum
Fissile Loading (kg Pu)	1350	2190	2740*
Breeding Ratio (wt % U^{235})	1.3	1.3	1.4 (1.3*)
Average Core Enrichment (% fissile Pu)	15	13	11 (12*)
Doppler Constant (T dk/dT)	-0.008	-0.0096	-0.008*
Primary System Configuration		Elevated Loop	
Number of Loops		3 Primary, 3 Secondary	
Primary Pump			
Flow (gpm)	38,500	77,000	62,800
TDH (ft)	379	355	432
Secondary Pump			
Flow (gpm)	45,300	87,000	55,600
TDH (ft)	226	250	256
Intermediate Heat Exchangers/Loop	1	2	1
Steam Generator Modules/Loop (Evaporator/Superheater/Reheater)	10/4/4	39/20/7	6/4/3

*Based on revised cross sections which include higher Pu^{239} α

a direct extrapolation of the AI Demonstration Plant, and would thus require a minimum of R&D beyond that planned for the Demonstration Plant. In keeping with this objective, both plants use the same fuel elements, control elements, fuel handling equipment, intermediate heat exchangers, and steam generator modules, except for numbers of units involved. The sodium pumps, valves, and reactor vessel and structure are components which would require extrapolation in size, but only a minimum of testing beyond that planned for the Demonstration Plant hardware. The ESADA study thus provided a measure of the potential LMFBR performance under a minimum R&D program.

Information generated under these programs, as well as AI's evaluation of the expected information from the AEC and foreign on-going programs, provided the starting point for Task I. The logic used in carrying out the Follow-On Study Program is shown in Figure I-1. The ESADA study and Task I effort were closely coordinated, to produce the Task I preliminary reference design (Block 1, Figure I-1). A preliminary capital cost estimate, a fuel cycle cost analysis, and an availability analysis were prepared for this design (Block 2), and compared with a light water reactor (LWR) competitive target plant (Block 3). Both plants were assumed to begin operation in the early 1980's, and were compared over a 20-year period on the basis of anticipated economic trends from an expanding combined LWR/LMFBR reactor economy.

The term "competitive" requires further definition. To be competitive, a new energy source should have sufficient economic advantage to provide for uncertainties in estimates, and to provide the necessary incentive for its application. It was assumed that, to be competitive, the LMFBR should have at least a 10% energy cost advantage (~0.4 mill/kwh) over a contemporary LWR. The economic assessment performed under the ESADA study indicated a 0.1 to 0.3 mill/kwh advantage for the Task I/ESADA design. Since this cost advantage did not meet the criterion, further upgrading of the plant design was required. Various potential improvements were identified and evaluated in the parameter survey (Block 4) and the alternate configuration studies (Block 5) performed in Task III. A preliminary assessment of these potential improvements, on the basis of the required R&D, led to a ranking of those improvements which were deemed achievable by the mid-1970's (Block 7), the period by which AI expects to market its first-generation commercial (1000-Mwe) plants.

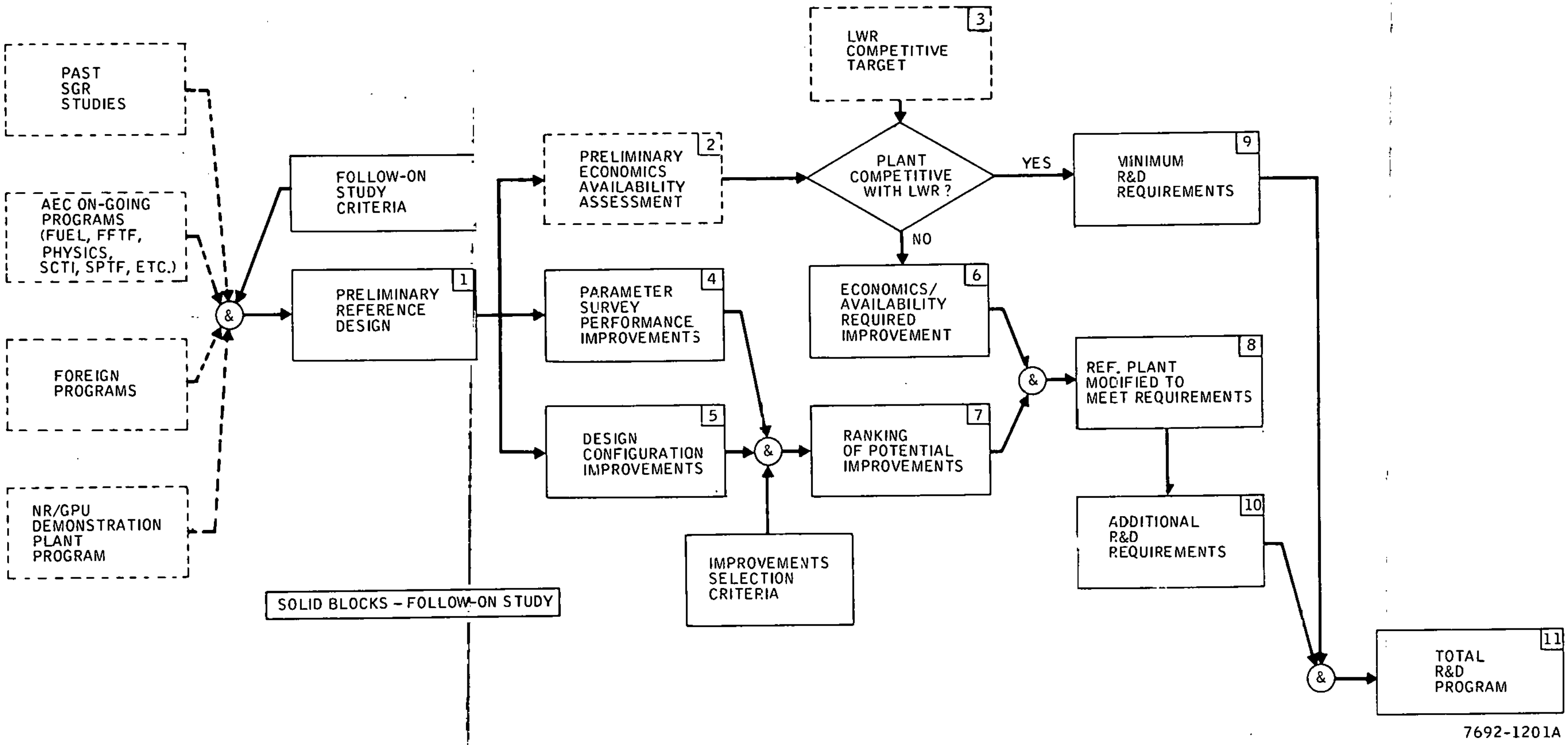


Figure I-1. Program Logic

Having established the criterion for competitiveness (~0.4 mill/kwh, relative to LWR's) and the necessary improvements to meet this criterion (e.g., 0.1 to 0.3 mill/kwh), the potential improvement list was reviewed. In order to allow a margin for the possibility of only partial achievement of improvement goals, and the possibility that LWR improvements will be greater than have been estimated, all the high-benefit improvements that were judged to be reasonably achievable, in time for use in the plant, were incorporated in the Task III final reference design (Block 8). The remaining steps involve the identification of the R&D for the Task I preliminary reference plant (Block 9), as well as the additional R&D required for the selected improvements (Block 10). This combined program defines the total R&D program required for the Task III final reference design, which will be described in Task IV (Block 11).

II. PLANT DESIGN BASES AND DESCRIPTION

A. PLANT DESIGN BASES

The established design bases for this study fall into four categories:

1) Study ground rules

2) Utility operational requirements

3) Safety requirements

4) Economic objectives.

The salient requirements falling under each of these categories are summarized in the following sections.

1. Study Ground Rules (Established by ANL)

1) The plant will be a 1000-Mwe (net) sodium-cooled fast breeder reactor, operating on the uranium-plutonium cycle.

2) The proposed plant will be of a design which will permit its initial commercial commitment within the 1975 to 1985 period. (AI has planned its program, based on sale in the early part of this period.)

3) The plant design optimization will reflect U.S. investor-owned utility financing conditions.

2. Utility Operational Requirements (Established by ANL and AI)

The plant shall be designed as a base load plant of low incremental power cost, but shall have provisions for following normal load changes.

3. Safety Requirements

The AEC General Design Criteria have formed the basis for the main safety guidelines and safety design features incorporated in the reference design. A summary of those guidelines most influential in the design follows. This summary is provided to highlight the influence of safety on the plant design.

a. Guideline No. 1

The reactor building complex will be designed to assure its mechanical integrity under maximum accident conditions, so as to meet the guidelines established in 10 CFR 100.

b. Guideline No. 2

The system design will provide highly reliable containment and control over liquid metals, to prevent or limit fire and other chemical interactions.

c. Guideline No. 3

Design provisions will be made to assure highly reliable and redundant emergency core cooling systems.

d. Guideline No. 4

The control and protective system will be sufficiently independent, redundant, and rapid to prevent significant damage to the core from all credible accidents.

e. Guideline No. 5

The core will be designed to provide an overall negative power coefficient at all operating conditions; the integrated Doppler effect from operating conditions to the onset of fuel damage will be sufficient to counteract all credible rapid reactivity perturbations; and the reactor and fuel elements will be designed to minimize interaction between fuel elements which could lead to significant fuel element-to-element failure propagation.

4. Economic Objectives (Established by AI)

The plant will be economically competitive with alternate energy sources, beginning operation in the early 1980's. As discussed previously, this was interpreted to mean that the proposed LMFBR design must show a levelized energy cost advantage of at least 10%, as compared to a contemporary LWR, over the initial 20 years of operation. This economic goal must be met with a design promising a mature plant "operating availability" of 92%. The corresponding "energy availability" goal for a three-loop plant is estimated at ~89%.

The design bases used in the Follow-On Study work are given in more detail in the Appendix. Certain of the specific bases merit discussion, because of the effect they have on the work.

a. Feed Fuel Composition

The contract ground rules specified a plutonium composition of 67 at.% Pu^{239}, 26 at.% Pu^{240}, 5 at.% Pu^{241}, and 2 at.% Pu^{242}. Typical LWR plutonium, which would likely be used for feed material in the early part of the plant lifetime, would have less Pu^{239} and more Pu^{241}. The effect of higher Pu^{241} content is to increase the reactivity swing of the core over the refueling cycle. This might result in a requirement for more control rods, to hold down the greater excess reactivity required, or could be compensated for by reducing the cycle time during the period LWR fuel is used as a feed.

b. Fuel Use Interest Rates

The contract ground rules call for a 10%/year charge rate. The possibility of fuel leasing is currently being considered by utilities and the financial community. This could have a significant effect on LMFBR optimization, as the charge for leased fuel would be very close to the prime money rate, or about 7 to 8%, in today's money market. This lower interest rate would weight inventory less heavily in core optimization.

c. Site

The site used as the basis for plant design was a modification of the AEC hypothetical site. Although none of the site conditions taken alone is unusual, from the standpoint of an actual site, it is unlikely that all these conditions would be present. These optimistic site assumptions, however, will not affect comparison with LWR's, because both plants would require similar features.

B. PLANT DESCRIPTION

The Follow-On Study plant has been designed as a load-following central-station power plant. Table II-1 summarizes the characteristics of the Task III final reference design. The characteristics of the Task I preliminary reference design are also given, to permit easy comparison of the two designs.

The major changes from the Task I preliminary reference design to the Task III final reference design, and their main effects, are summarized in the following listing.

1) Reactor outlet temperature has been increased from 1060 to 1140°F, with a corresponding improvement in steam cycle conditions and thermal efficiency. These lead to a significant capital cost reduction, with essentially no change in fuel cycle costs. While the higher outlet

	Task I Preliminary Reference Design	Task III Final Reference Design
OVERALL PLANT		
Nominal		
Reactor thermal power (Mwt)	2500	2400
Gross electrical power (Mwe)	1052	1040
Net electrical power (Mwe)	1004	1002
Plant efficiency (%)	40.7	41.7
Net heat rate (Btu/kwh)	8375	8175
Reactor outlet temperature (°F)	1060	1140
Primary loop ΔT (°F)	300	360
Steam generator Na inlet temperature (°F)	950	1070
Secondary loop ΔT (°F)	260	400
Steam Conditions		
Superheat/reheat temperature (°F)	900/900	1000/1000
Pressure (psig)	2400	2400
Condenser pressure (Hg abs)	1.5	1.5
Refueling Cycle (yr)	1/2	1
Reactor Containment	Double barrier	Double barrier
Fuel Material	Mixed PuO_2-UO_2	Mixed PuO_2-UO_2
Reactor Coolant	Sodium	Sodium
Reactor Geometry	Regular	Regular
Heat Transfer System Geometry	Loop	Loop
REACTOR		
Reactor Geometry		
Core height (in.)	50	43
Core L/D	0.54	0.42
Axial blanket length (in.)	12 (upper) 12 (lower)	12 (upper) 15 (lower)
Rows of radial blankets	2	2
Radial blanket height, inner/outer (in.)	62/50	56.5/43
Rows of reflector elements	1	1
Reactor subassemblies (total)	469	505
Fuel elements	256	274
Control rods	15	15
Shim-safety	9	9
Safety	6	6
Radial blankets	126	138
Inner	60	66
Outer	66	72
Reflectors	72	78
Fission gas plenum height (in.)	64	38
Inlet plenum height (in.)	36	36
Vessel OD (ft-in.)	22-9	29-0
Vessel height (ft) (bottom head to floor)	49	48
Vessel jacket ID (ft-in.)	24-11	31-8
Reactor cavity ID (ft-in.)	26-4	35-0
In-vessel fuel storage locations	91	0 (15 transfer positions)
Top shield type	Double rotatable plug	Double rotatable plug
Large plug diameter (ft-in.)	25-0	24-6
Small plug diameter (ft-in.)	13-6	15-6
Overall thickness (ft)	10.25	11.25
Steel plates (in.)	15	24
Serpentine (in.)	12	60
Steel top plate (in.)	15	15
Seal type	Double lip, He purge	Double lip, He purge

	Task I Preliminary Reference Design			Task III Final Reference Design		
REACTOR (Continued)						
Containment dome ID (ft-in.)	28-4			35-0		
Gas atmospheres						
Reactor cover gas	He			He		
Containment dome	N_2			N_2		
Reactor cavity	N_2			N_2		
Thermal-Hydraulics						
Thermal power (Mwt)	2500			2400		
Sodium outlet temperature (°F)	1060			1140		
Reactor ΔT (°F)	300 Average			360 Average		
Core ΔT (°F)	314 Average			371 Average		
Flow control	Fixed orifices			Fixed orifices		
Peak fuel pin power (kwt/ft) (100% power)	15			16		
Maximum nominal cladding ID temperature (°F)	1130			1221		
Hot spot cladding ID temperature (°F)	1200			1282		
Maximum heat flux (Btu/hr-ft^2)	785,000			690,000		
Average core heat flux (Btu/hr-ft^2)	495,000			437,000		
Cladding hot channel factor	1.16			1.14		
Radial peaking factor	1.26			1.22		
Axial peaking factor	1.25			1.23		
Reactor ΔP (psi)	100			87		
Coolant velocity (ft/sec)						
Maximum (at spacer)	32.8			-		
Average (maximum element)	27			27.5		
Reactor flowrate (lb/hr)	93×10^6			75.6×10^6		
Nuclear (Mid-Equilibrium) Characteristics						
Reactor power fractions						
Core	0.899			0.917		
Axial blankets	0.032			0.031		
Radial blankets	0.069			0.052		
Volume fractions (cell)	Fuel	Sodium	Metal	Fuel	Sodium	Metal
Core and axial blanket	0.328	0.479	0.193	0.439	0.326	0.235
Radial blanket, inner	0.550	0.298	0.152	0.556	0.281	0.163
Radial blanket, outer	0.577	0.292	0.131	0.584	0.269	0.147
Core fissile enrichments (at.%)						
Inner zone	11.4			10.5		
Outer zone	15.4			13.1		
Fissile masses (Pu + U-235) (kg)	2264			2744		
Core	1780			2191		
Axial blanket	116			176		
Radial blanket	368			377		
Initial loading (Pu + U) (kg)						
Core	13,339			19,090		
Axial blanket	7,463			13,132		
Radial blanket	15,365			15,688		
Breeding ratio	1.32			1.30		
Breeding ratio (including U-235)	1.28			1.27		
Core	-			0.94		
Average fuel burnup (Mwd/kgH)	75			67		
Doubling time (yr)	10			13		
Reactivity Coefficients						
Doppler constant (T dk/dT)	-0.0096			-0.0080		
Maximum radial sodium void ($)	6.82			8.37		
Total sodium void, core only ($)	6.03			8.07		
Isothermal sodium temperature (Δk/°F)	3.3×10^{-6}			4×10^{-6}		

	Task I Preliminary Reference Design	Task III Final Reference Design
REACTOR (Continued)		
Fuel and Blanket Elements		
Geometry	Hexagonal	Hexagonal
Lattice spacing (in.)	5.389	5.667
Distance across flats (external) (in.)	5.339	5.617
Housing wall (in.)		
Fuel element	0.130	0.140
Inner radial blanket	0.103	0.100
Outer radial blanket	0.091	0.100
Pins per subassembly/OD (in.)		
Fuel and axial blanket	217/0.25	217/0.30
Inner radial blanket	91/0.47	91/0.50
Outer radial blanket	61/0.58	61/0.62
Fuel and cladding material	Mixed PuO_2, UO_2 - Type 304 or 316 stainless steel	Mixed PuO_2, UO_2 - Type 316 stainless steel
Blanket and cladding material	Depleted UO_2 - Type 304 or 316 stainless steel	Depleted UO_2 - Type 316 stainless steel
Smeared fuel density (% theoretical)		
Fuel and axial blanket	80	85
Radial blanket	93	93
Pin spacers		
Fuel elements	Grid	Wire-wrap
Radial blanket elements	Wire-wrap	Wire-wrap
Fuel element length (in. overall)	214	180
Fuel (in.)	50	43
Axial blanket (each) (in.)	12	12 upper/15 lower
Gas plenum (in.)	64	38
Radial blanket element length (in. overall)	214	180
Inner blanket (in.)	62	56.5
Outer blanket (in.)	50	43
Gas plenum (in.)	10	10
Radial blanket burnup (Mwd/kgH)		
Inner	11.8	15.0
Outer	9.1	10.3
Control and safety rods		
Reactivity swing over equilibrium cycle ($)	5.0	3.5
Absorber material	Tantalum pins	Tantalum pins
Cladding	Type 304 stainless steel	Type 316 stainless steel
Thermal bond	NaK	NaK
Pins in array		
Shim-safety	16	-
Safety	19	-
Average rod worth ($)		
Shim-safety	1.6	1.2
Safety	2.0	2.4
Type of cooling	Thermal syphon	Thermal syphon
Type of drive	Electromechanical	Electromechanical
HEAT TRANSFER		
Number of Loops	3 primary and 3 secondary	3 primary and 3 secondary
Total Reactor Flow (lb/hr)	93×10^6	75.6×10^6
Reactor Inlet Temperature (°F)	760	780
Reactor Outlet Temperature (°F)	1060	1140
Pressure at Top of Reactor Vessel (psig)	1	1
Primary System Sodium Volume* (ft^3)	30,000	40,000
Primary Sodium Volume/Loop (ft^3)	6,000	6,700
Secondary Sodium Volume/Loop (ft^3)	14,000	13,800

*Includes three primary loops and reactor vessel

	Task I Preliminary Reference Design		Task III Final Reference Design	
HEAT TRANSFER (Continued)				
Primary Sodium Pumps*	3 (one/loop)		3 (one/loop)	
Type	Free-surface centrifugal		Free-surface centrifugal	
Flow (gpm)	77,000		62,800	
Head (ft)	355		432	
Brake horsepower (bhp)	6660		7000	
NPSH available (ft)	30		30	
Intermediate Heat Exchangers	6 (two/loop)		3 (one/loop)	
Type	Shell-and-tube unit		Shell-and-tube unit	
Duty/IHX (Btu/hr)	$1{,}425 \times 10^6$		$2{,}735 \times 10^6$	
Surface area/IHX (ft^2)	32,200		29,900	
Secondary Sodium Pumps*	3 (one/loop)		3 (one/loop)	
Type	Free-surface centrifugal		Free-surface centrifugal	
Flow (gpm)	87,000		55,600	
Head (ft)	250		256	
Brake horsepower (bhp)	5,400		4000	
Steam Generator Banks	3 (one/loop)		3 (one/loop)	
Evaporator duty/bank (Btu/hr)	$1{,}830 \times 10^6$		$1{,}610 \times 10^6$	
Evaporator surface area/bank (ft^2)	29,700		21,800	
Superheater duty/bank (Btu/hr)	510×10^6		670×10^6	
Superheater surface area/bank (ft^2)	12,800		11,200	
Reheater duty/bank (Btu/hr)	476×10^6		450×10^6	
Reheater surface area/bank (ft^2)	15,000		12,200	
Sodium inlet temperature (°F)	950		1070	
Sodium outlet temperature (°F)	690		670	
Steam Generator				
Superheater module				
Type construction	Shell-and-tube unit		Shell-and-tube unit	
Total number in plant	3 banks		3 banks	
Number of modules per bank	20		4	
Duty per module (Btu/hr)	25.5×10^6		167×10^6	
Surface area per module (ft^2)	640		2,800	
Tube data per module	79 tubes, 5/8-in. OD by 0.109-in. wall, 50-ft 6-in. effective length		380 tubes, 5/8-in. OD by 0.095-in. wall, 44-ft 11-in. effective length	
	Tube Side	Shell Side	Tube Side	Shell Side
Fluid	Steam	Secondary Na	Steam	Secondary Na
Flow per bank (lb/hr)	2,500,000	18,400,000	2,280,000	13,450,000
Temperature in (°F)	715	950	715	1070
Temperature out (°F)	905	860	1005	905
Outlet steam pressure (psia)	2515	-	2490	-
Design pressure (psi)	2700	200	2670	200
Design temperature (°F)	965	965	1085	1085
Material	2.25 Cr - 1 Mo	2.25 Cr - 1 Mo	Type 321 SS	Type 321 SS
Evaporator module				
Type construction	Shell-and-tube unit		Shell-and-tube unit	
Total number in plant	3 banks		3 banks	
Number of modules per bank	39		6	
Duty per module (Btu/hr)	46.8×10^6		268×10^6	
Surface area per module (ft^2)	762		3640	
Tube data per module	91 tubes, 5/8-in. OD by 0.095-in. wall, 51-ft effective length		380 tubes, 5/8-in. OD by 0.095-in. wall, 58-ft 7-in. effective length	
	Tube Side	Shell Side	Tube Side	Shell Side
Fluid	Steam/water	Secondary Na	Steam/water	Secondary Na
Flow per bank (lb/hr)	2,500,000	35,500,000	2,280,000	22,500,000
Temperature in (°F)	478	860	478	905
Temperature out (°F)	715	690	715	670
Inlet feedwater pressure (psia)	2600	-	2780	-
Design pressure (psi)	2800	200	2968	200
Design temperature (°F)	875	875	920	920
Material	2.25 Cr - 1 Mo	2.25 Cr - 1 Mo	2.25 Cr - 1 Mo	2.25 Cr - 1 Mo

*For 100% power condition

	Task I Preliminary Reference Design		Task III Final Reference Design	
HEAT TRANSFER (Continued)				
Reheater module				
Type construction	Shell-and-tube unit		Shell-and-tube unit	
Total number in plant	3 banks		3 banks	
Number of modules per bank	7		3	
Duty per module (Btu/hr)	68×10^6		149×10^6	
Surface area per module (ft^2)	2140		4060	
Tube data per module	161 tubes, 1-in. OD by 0.095-in. wall, 50-ft effective length		380 tubes, 1-in. OD by 0.058-in. wall, 41-ft effective length	
	Tube Side	Shell Side	Tube Side	Shell Side
Fluid	Steam	Secondary Na	Steam	Secondary Na
Flow per bank (lb/hr)	2,340,000	17,100,000	2,130,000	9,050,000
Temperature in (°F)	553	950	630	1070
Temperature out (°F)	905	860	1005	905
Outlet pressure (steam)(psia)	505	-	515	-
Design pressure (psi)	600	200	600	200
Design temperature (°F)	965	965	1070	1070
Material	2.25 Cr - 1 Mo	2.25 Cr - 1 Mo	Type 321 SS	Type 321 SS
TURBINE GENERATOR				
Type of Machine	Cross-compound		Tandem-compound	
Speed (rpm)	3600/1800		3600	
Main Steam Flow (lb/hr)	7.54×10^6		6.85×10^6	
Main Steam Pressure (psig)	2,400		2,400	
Main Steam Temperature (°F)TT	900		1000	
Reheat Steam Temperature (°F)TT	900		1000	
Reheat Steam Pressure (psig)	515		515	
Reheat Steam Flow (lb/hr)	7.0×10^6		6.4×10^6	
Condenser Pressure (in. Hg abs)	1-1/2		1-1/2	
Number of Feedwater Heaters	7		7	
Final Feedwater Temperature (°F)	478		478	
Gross Electrical Output (kw)	1,062,000		1,040,000	
Net Electrical Output (kw)	1,004,000		1,002,000	
Net Plant Heat Rate (Btu/kwh)	8375		8175	
Net Plant Efficiency (%)	40.8		41.7	
SUPPORT EQUIPMENT SEGMENT				
Refueling Cycle (mo)	6		12	
Elements Handled per Refueling (avg)				
Fuel	86		137	
Inner blanket (also rotate)	10		13	
Outer blanket (also rotate)	7		10	
Control rod	0.5		1	
Sodium Temperature (°F)	500		500	
SITE STRUCTURES AND SERVICES				
Reactor Building				
ID (ft)	176		140	
Inside height (cylinder) (ft)	105		121	
Hemispherical dome radius (ft)	88		70	
Overall height (ft)	193		191	
Type of containment	Double barrier		Double barrier	
Containment leak rates				
Inner barrier at 20 psi (%/day)	10		10	
Outer barrier at 10 psi (%/day)	0.5		0.5	
Atmospheres				
High bay	Air		Air	
Heat transfer vaults	N_2		N_2	
Seismic design ground accelerations (g)				
Design earthquake				
Horizontal	0.05		0.05	
Vertical	0.03		0.03	
Maximum hypothetical earthquake				
Horizontal	0.10		0.10	
Vertical	0.06		0.06	

Table II-1. 1000-Mwe LMFBR Follow-On Design Data temperature was estimated to reduce fuel burnup somewhat, this should be largely compensated for by the higher thermal efficiency.

2) The new core design features larger pins, a somewhat shorter core, higher reactor ΔT, and a continuous spiral-wire fuel pin spacer in place of a grid spacer. These lead to a significant reduction in fuel cycle costs, while facilitating annual refueling in lieu of a semi-annual refueling schedule (because of lower specific power and higher in-core breeding ratio). Annual refueling, when integrated with annual turbine maintenance, results in ~2% improvement in plant availability.

3) Spent fuel is removed from the reactor vessel in a programmed manner shortly after reactor shutdown, and stored in a sodium-filled vessel until ready for shipping. This capability eliminates the need for prolonged in-vessel storage until the next refueling, with the attendant fuel inventory penalties for such storage. This capability also permits an annual refueling schedule which coincides with annual turbine maintenance, and results in little or no economic penalty.

4) The straight shell-and-tube steam generator module configuration has been changed to a "hockey-stick" configuration. The new configuration preserves the many excellent features of the basic AI modular steam generator design, while removing its major disadvantage, limited capability (viz, of large modules to accommodate differential thermal expansion under severe transients). The new design permitted a factor of four increase in module size, which results in a significant reduction in capital costs. In addition, the higher reactor outlet temperature justified a substantial increase in the sodium inlet temperature to the steam generator, thereby requiring the replacement of 2.25 Cr - 1 Mo alloy with austenitic material for the superheater and reheater. The new temperature conditions also reduced the heat transfer surface requirement, resulting in a further significant reduction in capital cost.

5) The new design retains the three-loop system, but features a single IHX unit per loop, instead of two half-size units. This change,

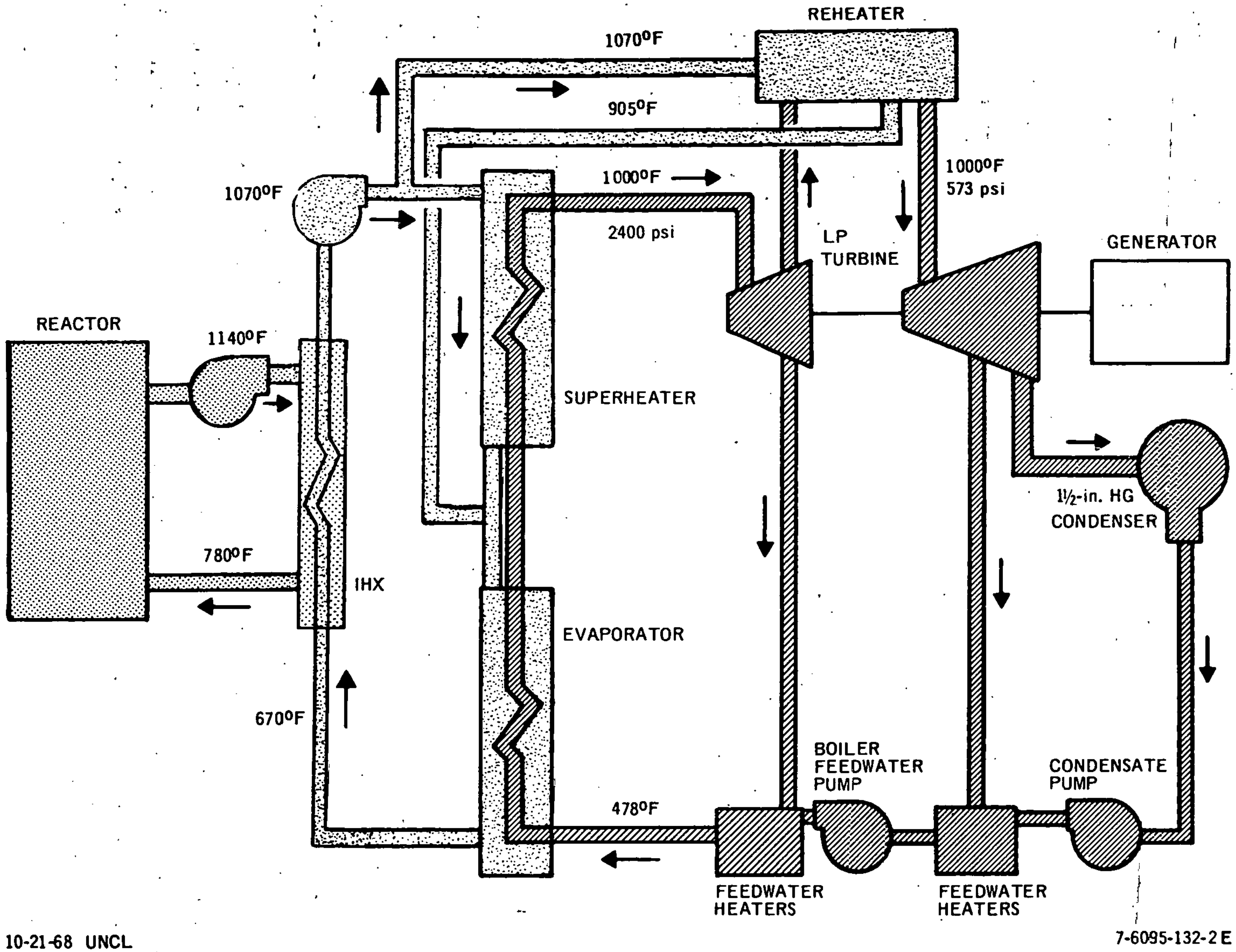


Figure II-1. Basic Flow Diagram

coupled with more efficient utilization of space, led to a reduction in the reactor building diameter, from 176 to 140 ft.

6) Because of the lower capital cost of the tandem-compound turbine generator, the selection was changed from a 3600/1800-rpm cross-compound turbine with 43-in. last-stage buckets to a 3500-rpm tandem-compound, six-flow turbine with 33-1/2 in. last-stage buckets.

The Task III final reference plant uses an elevated loop-type primary system configuration. The basic flow diagram, and values for the main process parameters, are shown in Figure II-1. Heat is transferred from the radioactive primary-sodium loop to the nonradioactive secondary-sodium loop for the production of steam in the steam generators. The primary-coolant system is composed of three identical heat-transfer loops, operating in parallel. The reactor vessel provides a common inlet and discharge for the primary loops. The primary system sodium coolant enters the reactor vessel at 780°F, absorbs energy from the fuel elements, and exits at 1140°F. The energy is transferred from the primary to the secondary system in the intermediate heat exchangers (IHX). The secondary system consists of three independent loops, one in series with each primary loop. The secondary sodium is pumped through the tube side of the IHX and the shell side of the steam generator. The secondary-system sodium coolant enters the IHX at 670°F and exits at 1070°F. Steam is generated in single-wall shell-and-tube heat exchangers of the modular type, at conditions of 2400 psig, 1000°F, with reheat to 1000°F. The turbine-generator complex is essentially the same as that used in fossil fuel plants.

1. Reactor

The reactor consists of a central cylindrical array of 274 fuel elements and 15 control rods (i.e., the core), surrounded by two rows of blanket elements and a row of stainless-steel reflector elements. The core fuel material is PuO_2-UO_2, and the blanket material is depleted UO_2. Tantalum control rods are used. The active core is 43 in. high, and has an equivalent diameter of 102 in.

The reactor is shown in elevation in Figure II-2. The reactor vessel houses the reactor core and blankets, provides containment for the sodium coolant, and provides transfer positions for new and spent fuel. The vessel is

Figure II-2. Reactor – Vertical Section (NO9620021) made of Type 304 stainless steel, is ~1 in. thick and 29 ft in diameter, and extends ~48 ft below the floor level. The vessel has three inlet and three outlet main-coolant-loop pipe penetrations, and one small inlet penetration for coolant supply to the low-pressure region below the reactor inlet plenum. Sodium enters the vessel at the same elevation as the outlet, flows downward to the high-pressure inlet plenum, and flows upward past the fuel and blanket elements. The elements are held down, against the upward flow of coolant, by a hydraulic holddown arrangement. Flow from the low-pressure region cools the control rods by a natural-convection balancing scheme. Three thermocouples are provided at the outlet of each fuel and blanket element.

Vessel internal structures consist of:

1) The lower reactor support assembly, which also serves as the high-pressure inlet plenum
2) Radial neutron-shield and reactor-restraint assemblies
3) Fuel handling transfer positions.

The lower reactor support assembly is made of two plates, perforated for the fuel and blanket element nozzles and control rods. Through tubes are provided at the control-rod locations, to allow flow from the low-pressure region for control-rod cooling. The core-subassembly nozzles seat on conical surfaces in the lower grid plate. All reactor vessel internals are designed to permit their removal through the opening resulting from removal of the small shield plug, should the need ever arise.

Helium is used for the reactor cover gas. Cover-gas pressure is maintained slightly above that of the reactor containment dome and operating area, to prevent in leakage. A cover-gas cleanup system is provided, to permit reactor operation with a small fraction (~1%) of fuel-pin cladding failures.

A double-rotatable shield plug forms the upper closure of the reactor vessel, and is an integral part of the fuel handling system. It provides radiation shielding for the operating area, a seal for the reactor cover gas, a heat-flow barrier, mechanical support for the control-rod drives, and the support and positioning mechanism for the in-core fuel handling machine. The thickness of the rotatable shield is ~11 ft. The plugs are made up of a steel top plate which

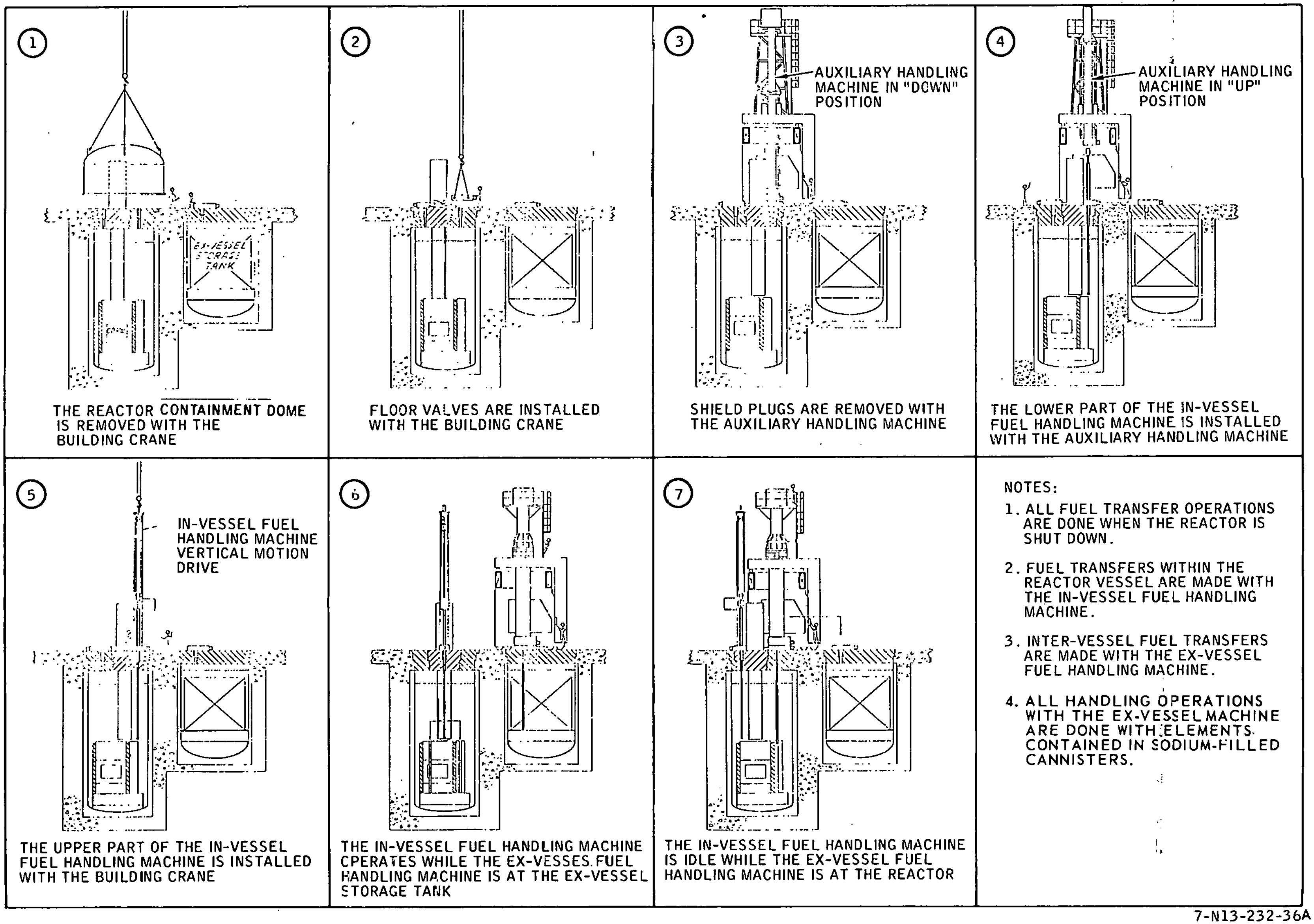


Figure II-3. In-Plant Fuel Handling Sequence is the basic structural member, crushed serpentine (hydrated magnesium silicate), and spaced stainless steel plates below the lower surface which act as a reflective thermal barrier. All penetrations are stepped, to prevent radiation streaming. Pans, supported from the underside of the plugs, are submerged in the sodium of the upper pool to minimize the interface between cover gas and sodium, thus inhibiting cover-gas entrainment.

An energy-absorber structure and reactor containment dome is provided over the top shield, to contain DBA reactions. The energy-absorbing structure, which is located directly above the top shield, limits the upward movement of the plugs and absorbs the kinetic energy imparted in the shield from the DBA. A missile barrier is provided to protect the containment dome. The containment dome prevents reaction products from being released to the reactor building. The dome area is vented to the equipment vaults through rupture discs.

The top shield consists of two rotatable plugs. The smaller rotatable plug is eccentric with respect to the larger. Rotation of the plugs permits positioning of the fuel handling equipment over any location. The reactor is refueled on a 1-yr cycle, with one-half of the core at an average discharge burnup of 67 Mwd/kgH (one-third of the core at 100 Mwd/kgH), one-fifth of the inner blanket, and one-seventh 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. In-vessel fuel handling is done under sodium. The fuel elements are placed in sodium-filled canisters while still in the reactor vessel. The ex-core fuel handling machine has an inert-gas atmosphere. Heat is removed by conduction to a NaK-cooled sleeve surrounding the machine.

The fuel handling procedure is shown in Figure II-3. Spent fuel is removed from the core and placed in a sodium-filled canister in the in-vessel transfer position and placed in the spent-fuel storage vessel by the ex-vessel handling machine. The fuel handling sequence is planned so that the blanket and lower-powered elements are handled first, thus allowing the higher-power elements to decay to a lower power level which can be handled without difficulty.

Secondary containment of primary sodium, in the event of a leak, is provided by the reactor vessel jacket. The vessel jacket provides a void volume which is less than that of the volume of sodium in the reactor vessel 2 ft above the downcomers of the outlet nozzles. This precludes uncovering the downcomers, should there be a sodium leak in the reactor vessel, and assures continued cooling through the main coolant loops. The vessel jacket also acts to protect against missile penetration of the cavity liner, if the reactor vessel ruptures as a result of the core disassembly accident, which is the Design Basis Accident (DBA). The containment dome above the shield and the reactor cavity liner provide part of the primary-containment barrier for radioactivity released from the reactor vessel, during and following the DBA.

The fuel element consists of a bundle of 217 fuel pins of 0.30 in. OD, contained inside a hexagonal Type 316 stainless steel wrapper tube. The fuel pins are made of pelletized fuel, with Type 316 stainless steel tubes as cladding. The pins are spaced laterally, on a triangular pitch of 0.36 in., by wire-wrap spacers, and they are vertically supported at their lower ends on support bars. The wrapper tube is 5.617 in. across flats outside, and is 0.140 in. thick. The fuel cladding is 0.0175 in. thick.

The fuel elements are positioned and supported at their lower end in the reactor-support assembly. The tubular lower end of the nozzle of the fuel element fits into a tube in the support assembly. Sodium flows into multiple openings around the nozzle, and upward through the fuel bundle. Coolant-flow distribution to the elements is controlled by varying the size and number of openings.

A gap is maintained between adjacent fuel elements, to provide for insertion and removal of the elements in the core array, to accommodate manufacturing tolerances in the fuel-element housings, and to allow for limited irradiation-induced structural material swelling and distortion. 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 flexible fingers mounted on the cylindrical stainless steel neutron-shield assembly. These flexible fingers act as cantilever beam springs that apply inward radial compressive forces. The elevation of the spacer pads and scalloped support fingers is established to assure negative core reactivity changes due to thermal bowing of the fuel elements.

This particular core clamping scheme does not adequately allow for large amounts of neutron-induced material swelling of the stainless steel core components, and their resultant bowing and distortion. Present data indicate that this swelling is in the range of 5 to 15 vol %.

To take material swelling into account, further design work on the core clamping device and the core components is necessary. In addition, tests are necessary to more accurately define the magnitude of material swelling, and possibly to find materials which are resistant to it.

Reactivity is controlled by vertical movement of control rods, containing a neutron absorber (tantalum), in and out of the core. 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 at any time.

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 grid-plate 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 released by de-energizing the latch magnet, allowing the absorber assembly 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.

2. Sodium Heat-Transfer System

Three one-third capacity heat-transfer circuits are provided. The intermediate heat exchangers are elevated, so that core decay-heat removal is assured by natural-convection circulation through any one of the three loops;

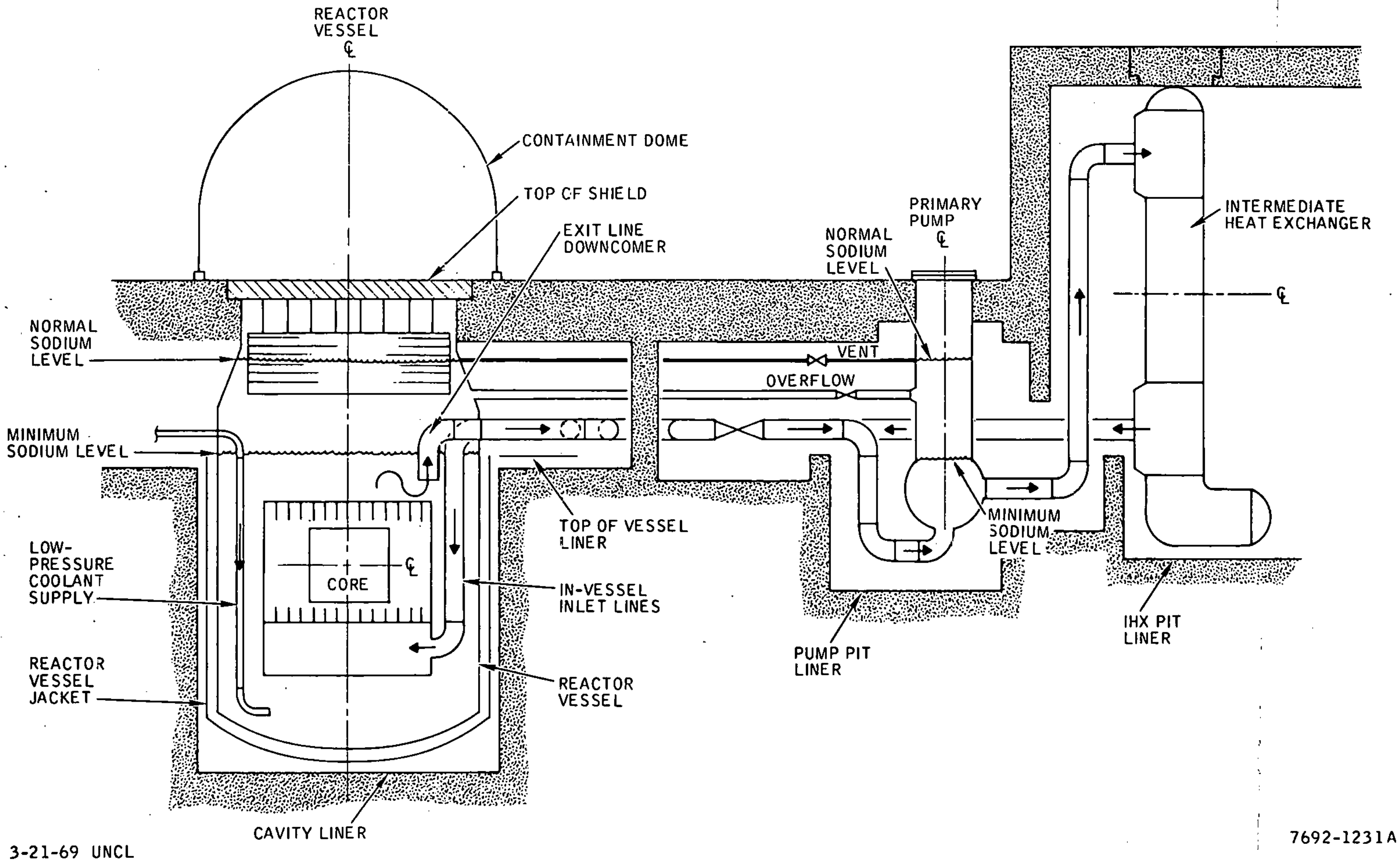


Figure II-4. Reactor and Primary Loop Elevations and uncovering of the core, as the result of a heat-transfer system leak, is prevented. Figure II-4 shows the relative elevations of loop components. In the event of a component failure in one of the loops, the reactor can be operated at a reduced power on two of the three circuits, with the third isolated to allow decay of sodium activity. One-loop operation is not permitted.

The sodium pumps are variable-speed free-surface single-stage centrifugal units. The main primary loops contain loop-isolation valves, a flow-control butterfly valve to control thermal-convection flow following a reactor trip, and a check valve to prevent backflow from the high-pressure plenum on loss of a pump. The IHX's are counterflow hockey-stick shell-and-tube heat exchangers, with the primary sodium on the shell side.

All primary-loop components are located in nitrogen-filled shielded equipment vaults. Except for the piping to the IHX, the secondary-system components are located in the steam-generator building.

The steam generators are divided into three banks, one for each secondary loop. Each bank is arranged separately in an in-line array with four superheat modules, three reheater modules, and six evaporator modules. The modules are mounted vertically, and are shell-and-tube heat exchangers with a hockey-stick configuration. Steam is on the tube side. All the modules are similar in design, having 380 tubes and varying only in the length and diameter of the tubes.

Each module has a hydrogen detector mounted on the sodium-outlet piping, to detect any water or steam leakage into the sodium. It is expected that any leak normally will be detected in time to shut down the affected bank before significant damage is done. Rupture discs and a relief system are provided, in the event of a failure large enough to cause excessive pressures. The relief system separates liquid sodium from the gases, and holds the sodium in a dump tank. The gases are discharged to the atmosphere through separators and a vent stack.

3. Turbine Generator Plant

The turbine generator is a 3600-rpm, tandem-compound, six-flow, 33-1/2 in. Last Stage Bucket (LSB) unit. The rated capability is 1040 Mwe at 0% makeup and 1.5 in. Hg abs. Steam conditions are 2400 psig/1000°F at the throttle, with 1000°F reheat. The heat balance diagram is shown in Figure II-5.

4. Plant Layout and Buildings

The overall plant arrangement is shown in Figure II-6. The reactor building, shown in Figures II-7 and II-8, is conventional reinforced concrete, with a steel liner to control leakage. The building houses the reactor, the fuel handling equipment, and all portions of the primary-sodium system including the intermediate heat exchangers. Containment to protect against the accidental release of radioactivity is provided by the building liner and the equipment vaults which enclose the reactor and primary system.

As shown in Figure II-6, the steam generators are located in a separate conventional steel building. This building also contains the steam-generator sodium-side pressure-relief equipment and secondary-sodium pump and surge tank. The turbine-generator building houses the prime mover and its associated equipment.

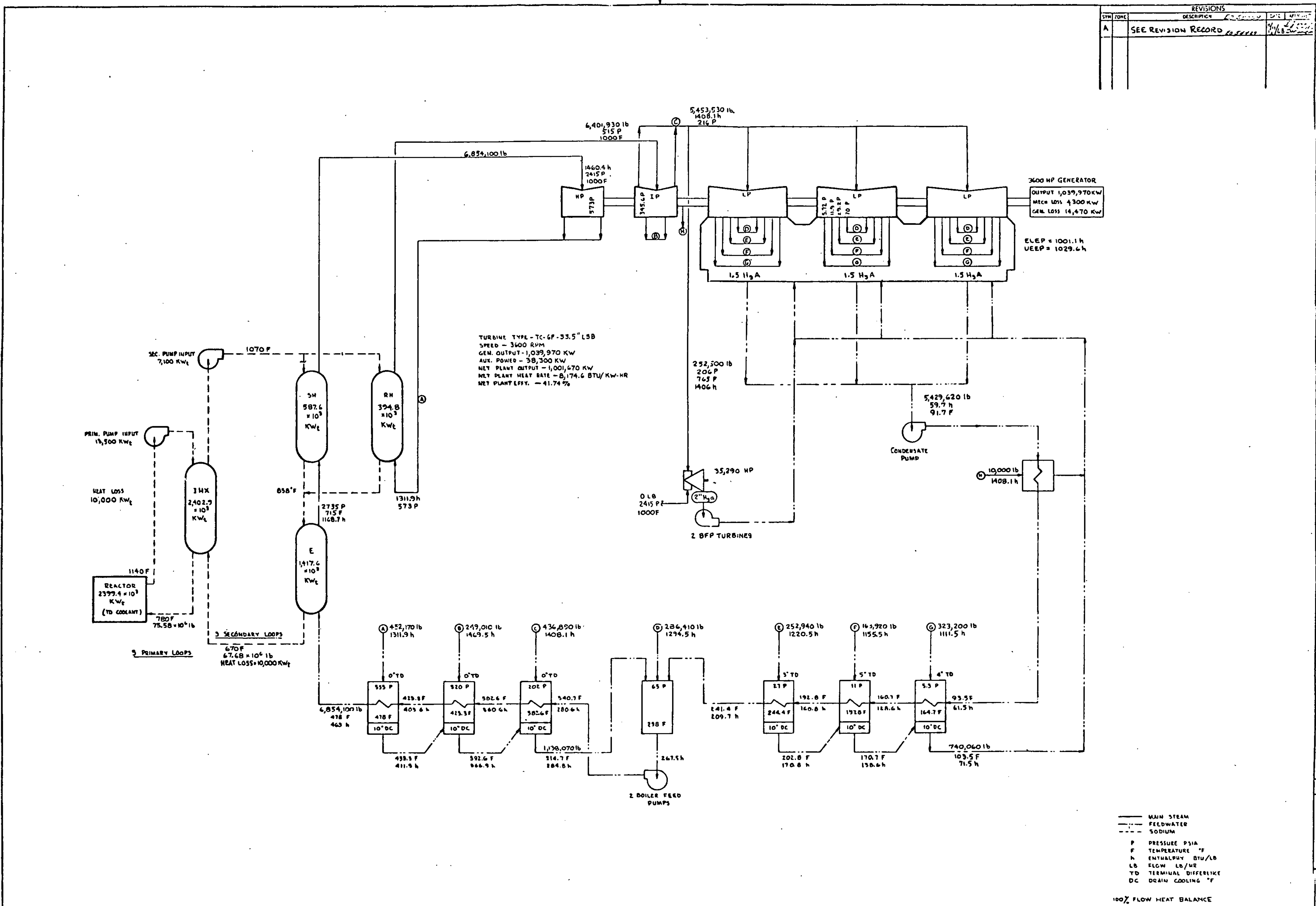


Figure II-5. Heat Balance Diagram (NO9630009)

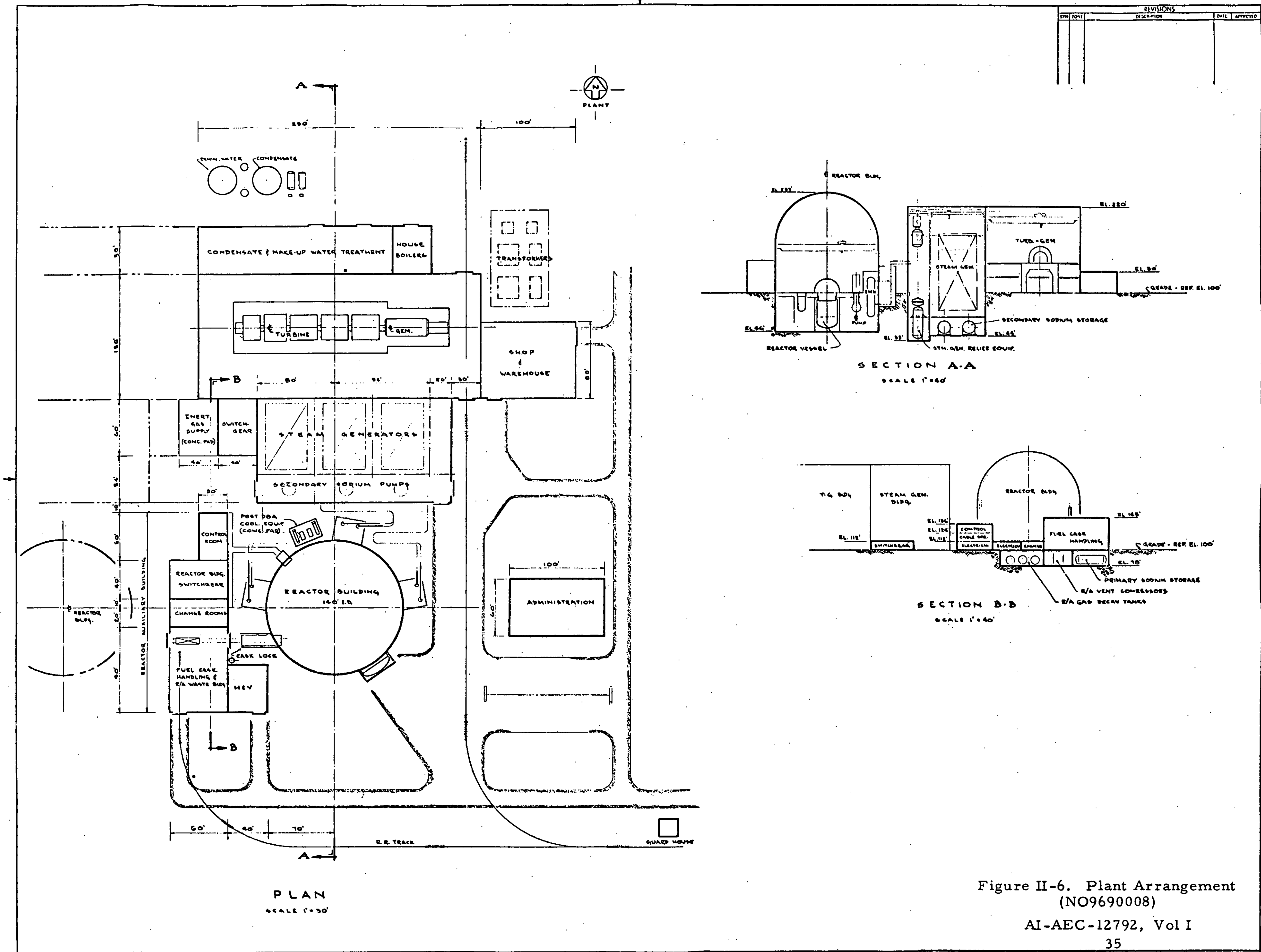
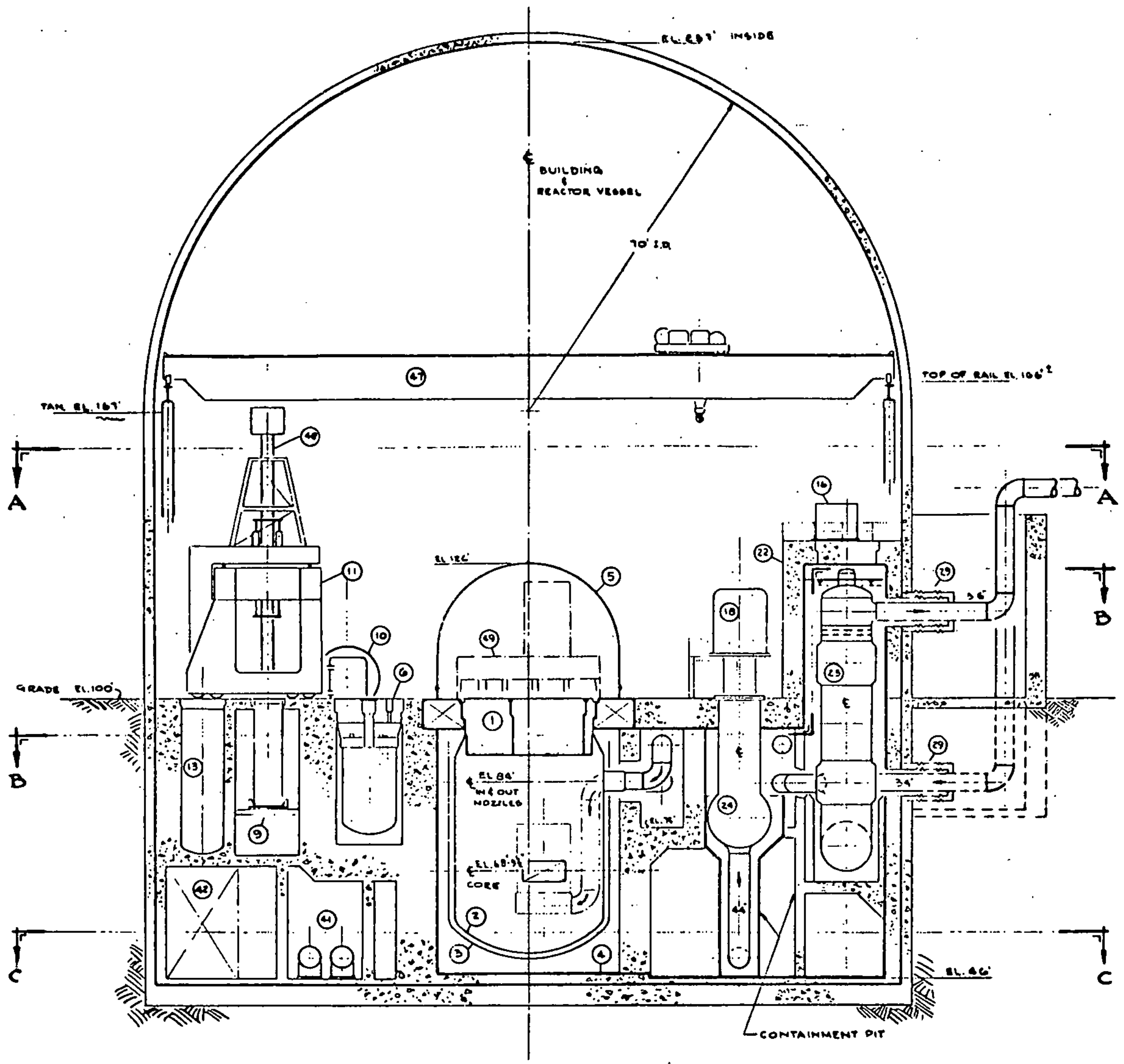
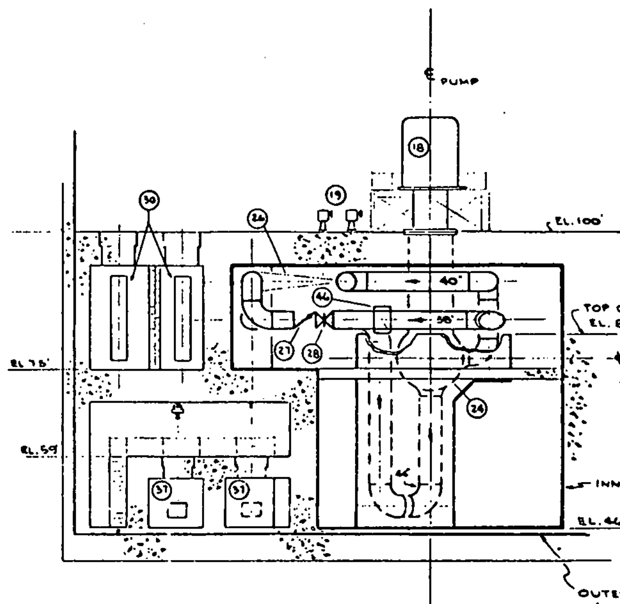


Figure II-6. Plant Arrangement (NO9690008)

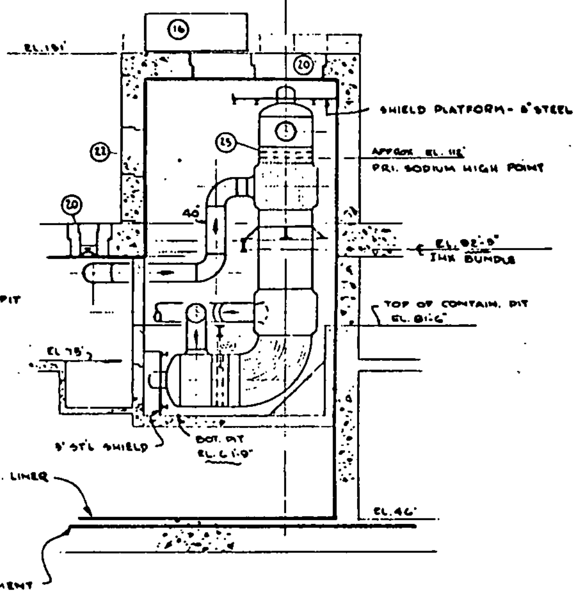
REVISIONS				
SYM	ZONE	DESCRIPTION	DATE	APPROVED



SECTION D-D



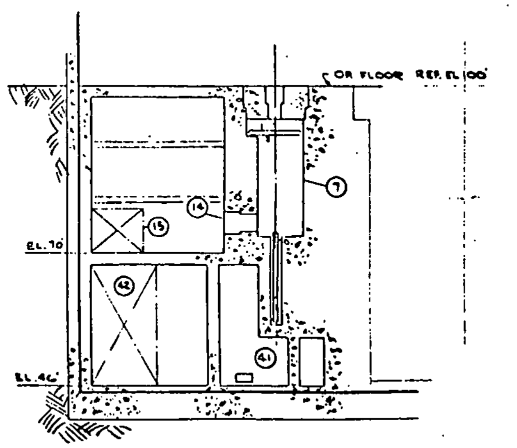
SECTION F-F



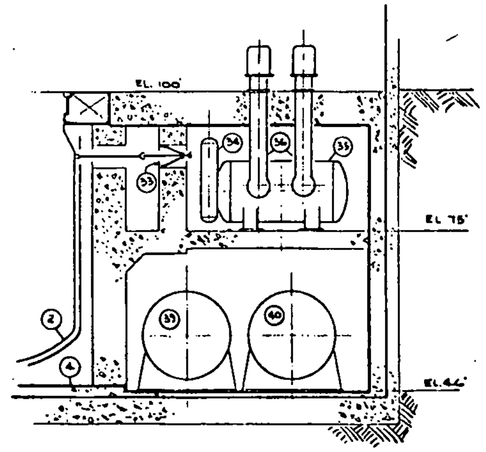
SECTION G-G

EQUIPMENT

1. REACTOR TOP SHIELD
2. REACTOR VESSEL
3. REACTOR VESSEL LINER
4. CAVITY LINER
5. REACTOR CONTAINMENT DOME
6. SPENT FUEL STORAGE VESSEL
7. FUEL CANISTER ENCAPSULATING CELL
9. SHIPPING CASK
10. SHIPPING CASK AIR LOCK
11. FUEL HANDLING GANTRY
13. PROCESS EQUIP. WASH CELL
14. VIEWING WINDOW
15. COOLING EQUIPMENT FOR SPENT FUEL STORAGE VESSEL
16. RECIRCULATING H&V UNITS
18. PRIMARY PUMP DRIVE
19. VALVE OPERATOR
20. ACCESS PLUG FOR IHX MAINTENANCE
22. IHX ENCLOSURE
24. PRIMARY PUMP
25. IHX
26. PRIMARY BLOCK VALVE
27. CHECK VALVE
28. THROTTLE VALVE
29. SECONDARY PIPING PENETRATION SEAL
30. COLD TRAP
33. ATMOSPHERE SEAL
34. REACTOR OVERFLOW TANK
35. DEGASIFIER/SURGE TANK
36. DEGASIFIER RETURN PUMPS
37. PRIMARY Na SERVICE PUMP
39. PRIMARY Na DRAIN TANK - 53,000 GAL.
40. PRIMARY Na DRAIN TANK - 30,000 GAL.
41. R/A LIQUID WASTE TANKS & PUMPS - DRAINS FROM WASH & ENCAPSUL. CELLS
42. COVER GAS PURIFICATION SYSTEM
46. PRIMARY FLOWMETER
47. BUILDING CRANE - 120 TON CAPACITY
48. EX-VESSEL FUEL HANDLING MACHINE
49. TOP SHIELD HOLD-DOWN STRUCTURES



SECTION E-E



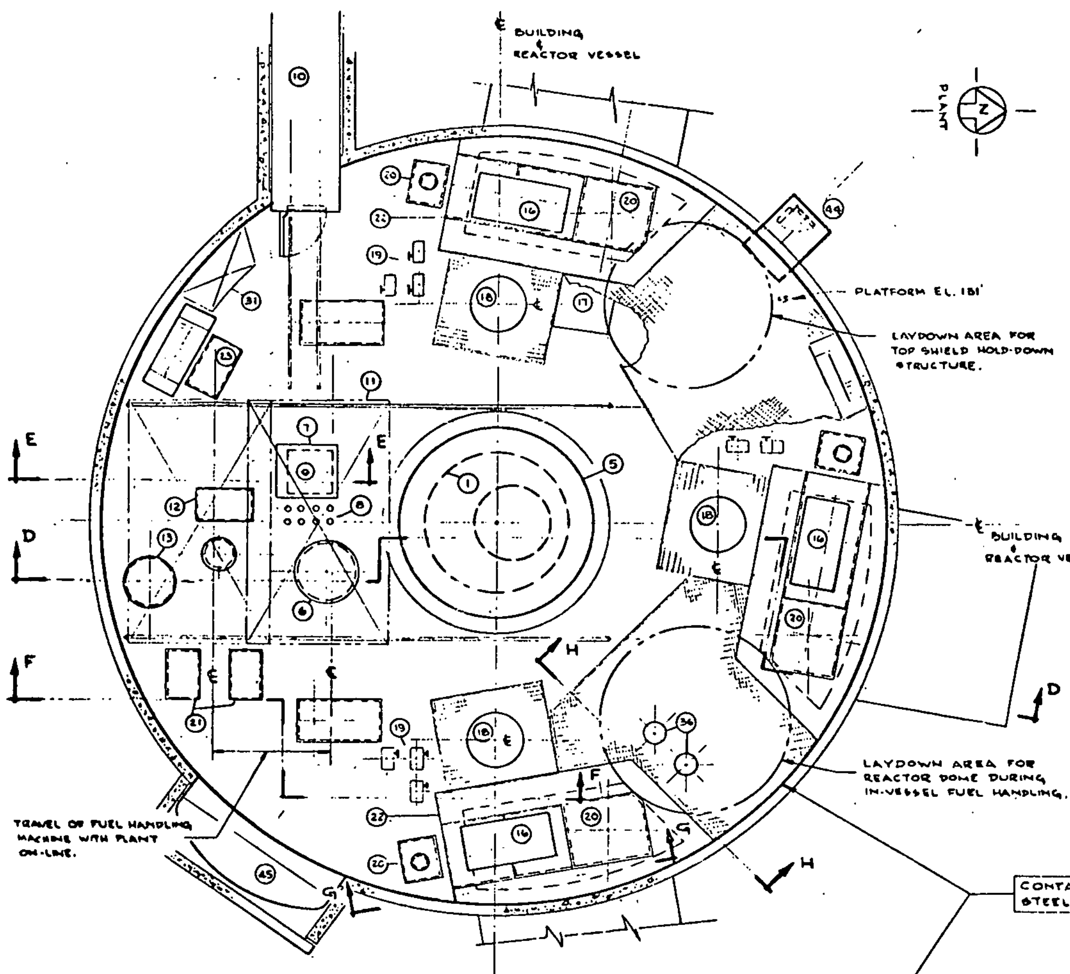
SECTION H-H

FOR PLAN VIEWS SEE—
DWG No. NO9696007

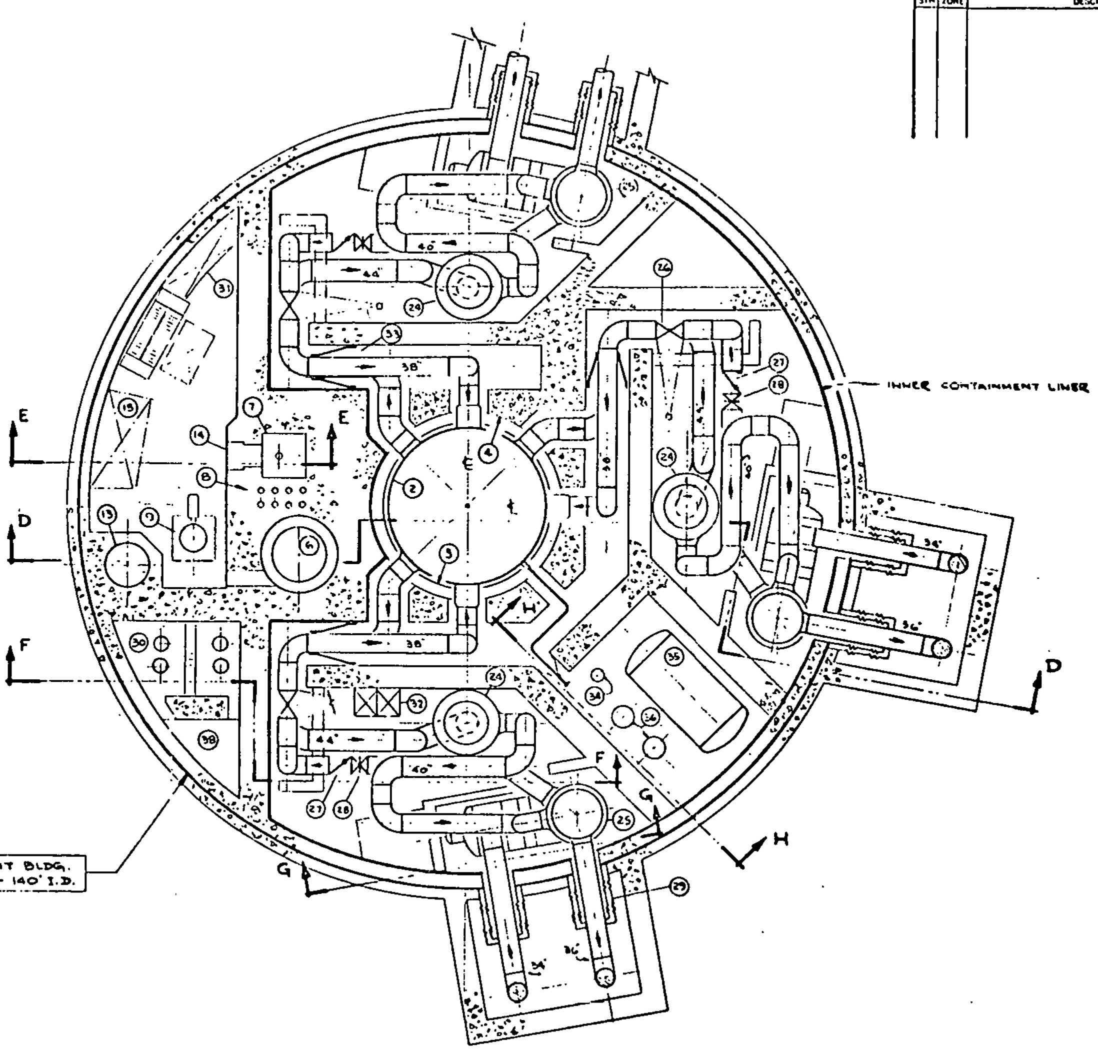
Figure II-7. Reactor Building Equipment Arrangement Sections (NO9696008)

NO9696008

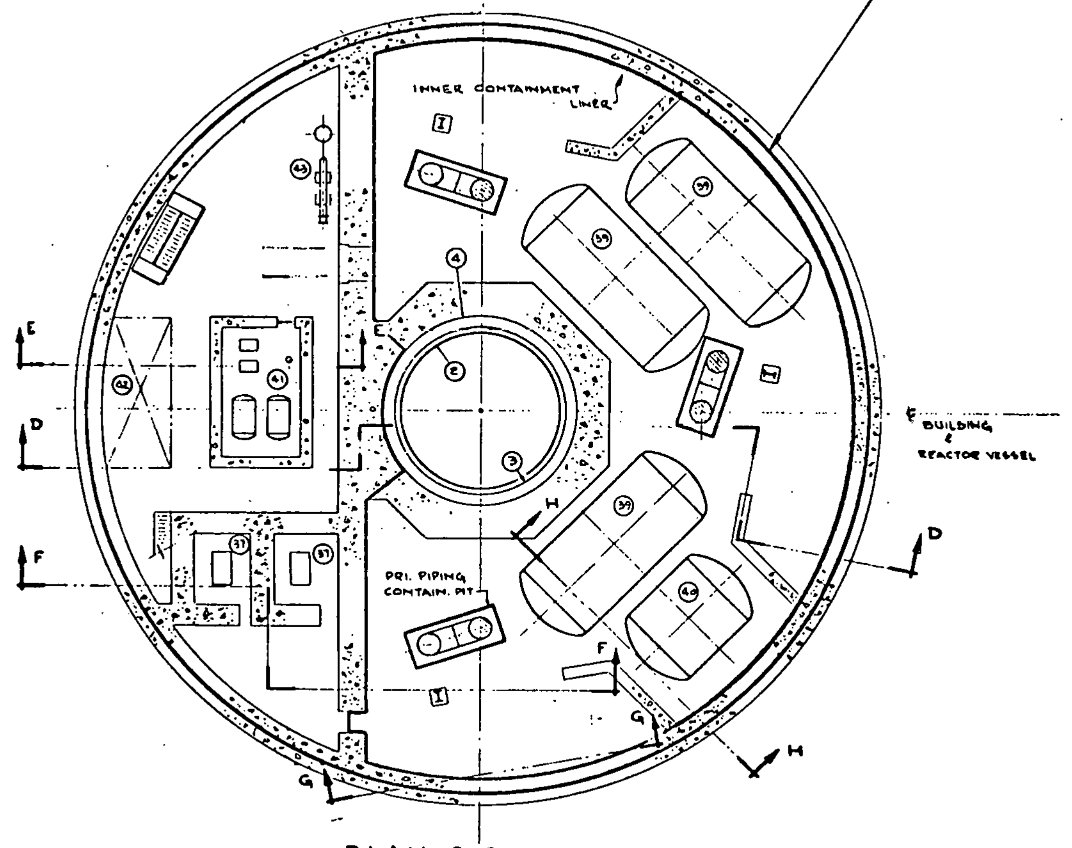
REVISIONS				
SYM	ZONE	DESCRIPTION	DATE	APPROVED



PLAN A-A - ABOVE OPERATING FLOOR



PLAN B-B - BELOW OPERATING FLOOR



PLAN C-C

EQUIPMENT

1. REACTOR TOP SHIELD
2. REACTOR VESSEL
3. REACTOR VESSEL LINER
4. CAVITY LINER
5. REACTOR CONTAINMENT DOME
6. SPENT FUEL STORAGE VESSEL APPROX. 200 STORAGE POSITIONS - 'LAZY SUZAN' STORAGE
7. FUEL CANISTER ENCAPSULATING CELL.
8. REACTOR COMPONENT WASH CELL AND MISC. STORAGE PITS
9. SHIPPING CASK
10. SHIPPING CASK AIR LOCK - 8'×10' CLEAR
11. FUEL HANDLING GANTRY
12. PLUG FOR BELOW-FLOOR MAINTENANCE OF FUEL HANDLING MACHINES.
13. PROCESS EQUIP. WASH CELL - 8' DIA - SIZED BY PRI. PUMP
14. VIEWING WINDOW
15. COOLING EQUIPMENT FOR SPENT FUEL STORAGE VESSEL
16. RECIRCULATING H & V UNITS
17. PRE-AMP ROOM
18. PRIMARY PUMP DRIVE
19. VALVE OPERATORS
20. ACCESS PLUGS FOR IHX MAINTENANCE
21. COLD TRAP ACCESS PLUGS
22. IHX ENCLOSURE
23. EQUIP. ACCESS HATCH (THROUGH TO BOT. LEVEL)
24. PRIMARY PUMP
25. IHX
26. PRIMARY BLOCK VALVES
27. CHECK VALVE
28. THROTTLE VALVE
29. SECONDARY PIPING PENETRATION SEALS (DOUBLE BELLOWS)
30. COLD TRAPS
31. H & V AND SERVICE PIPE CHASE
32. PRESSURE RELIEF PANELS FOR VAULT RELIEF TO DRAIN TANK BELOW (DBA RELIEF) - TYPICAL EACH VAULT
33. ATMOSPHERE SEAL - TYPICAL
34. REACTOR OVERFLOW TANK
35. DEGASIFIER / SURGE TANK
36. DEGASIFIER RETURN PUMPS
37. PRIMARY Na SERVICE PUMPS
38. PRIMARY Na SERVICE PIPE CHASE
39. PRIMARY Na DRAIN TANK - 53,000 GAL
40. PRIMARY Na DRAIN TANK - 30,000 GAL
41. R/A LIQUID WASTE TANKS - DRAINS FROM WASH & ENCAPSUL. CELLS
42. COVER GAS PURIFICATION SYSTEM
43. POST DBA COOLING EQUIP. - USED DURING NORMAL OPERATION - EMERGENCY EQUIPMENT IS OUTSIDE BUILDING
44. PERSONNEL AIR LOCK - 5'6" × 7' CLEAR - AT GRADE
45. LARGE EQUIPMENT ACCESS OPENING - 32' DIA. - WILL CLEAR REACTOR TOP SHIELD AND IHX

FOR SECTIONS SEE DWG. No. N09696008

Figure II-8. Reactor Building Equipment Arrangement Plans (NO9696007)

III. ENGINEERING CONSIDERATIONS

A. PLANT DESIGN STUDIES

In the plant studies, the Task I preliminary reference design characteristics were systematically investigated to determine:

1) Design improvements to be incorporated in a new, more advanced, reference design for Task III

2) The economic worth of potential improvements to be used in the evaluation of the Task IV R&D recommendations.

The economics of the Task I preliminary reference design were previously evaluated, as part of a study performed by AI for the Empire State Atomic Development Associates (ESADA). Although power costs computed in the ESADA studies were less than for contemporary light water reactor systems, the 10% margin considered necessary, in order to allow for uncertainties and to provide the incentive for large-scale application of LMFBR's, was not achieved.

Improvements in the preliminary reference design, to achieve the desired cost margin, were considered in the Task III plant design studies. These improvements fall into the areas of design, configuration, material selection, and changes in design parameters (i. e., temperature levels, fuel pin sizes, etc.). The complete list of the improvements considered is shown in Table III-1.

The approach taken, in carrying out the plant design studies and selecting the Task III final reference design, was to perform a systematic perturbation of variables and investigation of design options, based on the Task I preliminary reference design. The constraints, requirements, assumptions, independent variables, and design options were first established, to define the limits of the study. Reactor core designs were developed, covering the range of core design parameters. Based on the fuel cycle costs determined in the core parameter studies and capital cost data, the plant system parameters were studies. The results of the core and system parameter studies were then combined and evaluated, to select the plant characteristics for the Task III final design work.

The overall economics in the parameter studies were measured by the incremental cost variation in those plant elements affected by the changes ("partial

TABLE III-1

POTENTIAL IMPROVEMENTS CONSIDERED DURING TASK III

DESIGN POINT CHANGES
- OUTLET TEMPERATURE
- FUEL LINEAR POWER
- PRIMARY/SECONDARY ΔT
- PRIMARY ΔP
- FUEL BURNUP
- MAXIMUM COOLANT VELOCITY
- STEAM CONDITIONS

PLANT PARAMETER CHANGES
- REFUELING INTERVAL
- FUEL PIN SIZE
- FUEL CLAD THICKNESS
- REACTOR BUILDING SIZE
- FUEL SMEAR DENSITY
- NUMBER OF COOLANT CIRCUITS
- NUMBER OF IHX's PER LOOP
- CORE ENRICHMENT ZONES
- STEAM GENERATOR MODULE SIZE
- AMOUNT OF EX-VESSEL STORAGE IN BUILDING
- IHX SODIUM VELOCITY
- BLANKET LENGTH AND RESIDENCE TIME

CONFIGURATION CHANGES
- FUEL ELEMENT SPACER TYPE (WIRE WRAP vs GRID)
- SUPERHEATER MATERIAL (AUSTENITIC vs FERRITIC)
- ELIMINATION OF VALVES
- FUEL HANDLING – DIRECT FUEL REMOVAL FROM STORAGE
- BASE LOAD OPERATION
- ALTERNATE CLAD MATERIAL

DESIGN CHANGES
- VENTED vs NON-VENTED FUEL PIN
- ON-LINE FUEL REMOVAL
- BELLOWS TYPE IHX
- STEAM GENERATOR TURBULATOR INSERT
- TURBINE (CROSS COMPOUND vs TANDEM COMPOUND)
- FUEL ELEMENT OPEN HOUSINGS
- VARIABLE BLANKET ORIFICES
- STEAM GENERATOR TUBE JOINT – EXPLOSIVE BOND vs WELDED
- BACKUP SHUTDOWN DEVICE/EMERGENCY COOLING SYSTEMS – ELIMINATION OF PRESENT DBA
- ELIMINATION OF OUTLET THERMOCOUPLES
- ELIMINATION OF COVER GAS CLEANUP

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power costs," in mills/kwh). The principal items varied were capital costs, fuel cycle cost, and plant availability. The independent variables were then evaluated over the range of interest; and, in general, the variation was by perturbation of the Task I preliminary reference design. Each perturbation was studied separately, using a variation of one independent parameter at a time while holding all others constant. The sum of the economic changes was then taken as an approximation of the total economic change which would be incurred, if all changes were made. This assumed noninteraction of all of the variables involved; however, in many cases, the interaction was important enough that additional study was required.

Two analytical models for the effect of temperature on the maximum allowable fuel burnup were utilized. One assumed no temperature effect on achievable fuel burnup (constant burnup model), and the other was an engineering estimate of the maximum likely temperature effect on burnup (variable burnup model). Lacking significant data or background to judge otherwise, the design selection was based on the average of these two models. Sensitivity studies of key parameters (e.g., fabrication and equipment costs) were considered, but they did not affect the optimum design points.

The requirements, constraints, and assumptions needed to begin the Task III plant design studies were based on the overall plant requirements, and are shown in Table III-2. The list of improvements previously shown in Table III-1 was then limited, based on engineering judgement, to those which appeared to be most effective for the 1000-Mwe Follow-On Program, and which met the requirement that research and development work could be completed in time for a plant commencing operation in the early 1980's. The list of independent variables and design optimum selected to guide the studies is shown in Table III-3.

To allow a meaningful evaluation of core parameter studies, it was necessary to adjust the results to account for the effect of core design changes on capital cost, availability, pre-startup Pu inventory, fuel burnup, cladding thickness, and sodium pumping power. The core design parameter variations and the results of the core parameter studies are summarized in Table III-4. The evaluation of these results led to the following conclusions:

1) There is a strong incentive for a core with high fuel volume fraction.

2) The most economical way to obtain high fuel fraction is by the use of larger pins and smaller pin pitches.

3) Economics are relatively insensitive to fuel pin linear power.

4) Increasing fuel burnup reduces fuel cycle costs.

Figure III-1 summarizes the most important results of the studies, relative to optimization of system operating parameters. This figure shows the effect of the burnup model assumptions on indicated optimum design points. The conclusions of the system parameter studies were:

1) The best (minimum cost) outlet temperature was between 1060 and 1200°F, and depends on the burnup model used.

2) The minimum cost design is sensitive to the burnup model.

3) Austenitic superheater material, if not subject to stress corrosion problems, is economic in the outlet temperature range of interest.

TABLE III-2

REQUIREMENTS, CONSTRAINTS, AND ASSUMPTIONS FOR TASK III PLANT DESIGN STUDIES

PERFORMANCE

- 1000-Mwe NET ELECTRICAL
- ~2500 Mwt

OPERATION

- 90% AVAILABILITY
- SWING LOAD OPERATION
- REFUELING EITHER 6 months OR 1 year INTERVALS
- DIRECT FUEL REMOVAL

SAFETY

- NEGATIVE POWER COEFFICIENT
- CONTAIN MAJOR CORE EXCURSION
- ADEQUATE CONVECTION FLOW
- ELEVATED LOOP
- DOUBLE BARRIER CONTAINMENT
- LIMITED ROD WORTH

DESIGN

- 217 PINS PER ELEMENT (DIRECT FUEL REMOVAL LIMIT)
- TWO ROWS RADIAL BLANKETS
- DOUBLE ROTATING PLUG FUEL HANDLING
- TANTALUM CONTROL
- MODULAR STEAM GENERATOR
- LIMIT CLAD STRESS (ADJUST THICKNESS AND PLENUM HEIGHT)
- 75 Mwd/kgH FUEL BURNUP CAPABILITY WITH 1200°F CLAD HOT SPOT TEMPERATURE
- PLANT SITE PER TID-7025 WITH MODIFICATION
- DESIGN LIFE OF 30 YEARS

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TABLE III-3

Independent Variables and Design Options Studied for Task III

INDEPENDENT VARIABLE	RANGE
• FUEL PIN SIZE	0.200 to 0.30 in.
• CORE HEIGHT	18 to 50 in.
• CORE OUTLET TEMPERATURE	800 to 1200°F
• PRIMARY ΔT	150 to 470°F
• SECONDARY ΔT	105 to 490°F
• FUEL ELEMENT BURNUP (AVERAGE)	60 to 120 Mwd/kgH
• FUEL SMEAR DENSITY	72 to 88%
• FUEL LINEAR POWER	12 to 20 kw/ft
• STEAM CONDITIONS	
▲ PRESSURE	1200 to 2400 psig
▲ TEMPERATURE	600 to 1000°F
• AXIAL BLANKET LENGTH	9 to 15 in.
• RADIAL BLANKET RESIDENCE TIME	2 to 7 yr

DESIGN OPTIONS

- FUEL PIN SPACER (GRID vs WIRE WRAP)
- SUPERHEATER MATERIAL (FERRITIC vs AUSTENITIC)
- TURBINE TYPE (CROSS COMPOUND vs TANDEM COMPOUND)
- IHX UNITS PER LOOP (ONE vs TWO)
- NUMBER OF HEAT TRANSFER CIRCUITS (TWO vs THREE)
- STEAM GENERATOR MODULE (79 to 3280 TUBES)
- SPENT FUEL STORAGE
- PLANT ARRANGEMENT

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TABLE III-4

RESULTS OF CORE PARAMETER STUDY FUEL CYCLE COST ADJUSTMENTS

1	2	3	4	5	6	7	8	9	10	11	12
Case	Fuel Pin Diameter (in.)	Reactor ΔT (°F)	Maximum Linear Pin Power (kw/ft)	Maximum Coolant Velocity (ft/sec)	Core Height (in.)	Discharge Burnup (Mwd/kgH)	Smear Density (% TD)	Pin Pitch (in.)	Cost Adjustments (mills/kwh)	Uncorrected Fuel Cycle Cost (mills/kwh)	Adjusted Costs (mills/kwh)
1	0.250	300	15.3	32.8	50	75	80	0.342	-	0.904	0.904
2	0.200	x	x	x	x	x	x	0.308	(0.001)	1.151	1.150
3	0.300	x	x	x	x	x	x	0.378	0.035	0.804	0.839
4	x	260	x	x	x	x	x	0.353	(0.014)	0.938	0.924
5	x	360	x	x	x	x	x	0.328	0.070	0.863	0.933
6	x	x	12	x	x	x	x	0.325	(0.031)	0.907	0.876
7	x	x	20	x	x	x	x	0.363	0.080	0.936	1.016
8	0.300	x	20	x	x	x	x	0.399	0.083	0.782	0.865
9	x	360	20	x	x	x	x	0.348	0.129	0.854	0.983
10	x	x	20	x	35.6	x	x	0.343	0.066	0.923	0.989
11	x	x	x	40	x	x	x	0.328	0.027	0.887	0.914
12	x	x	x	x	35.6	x	x	0.324	0.003	0.910	0.911
13	x	x	15.4	30.0*	18	x	x	0.287	0.066	1.091	1.157
14	x	x	x	14.3	18	x	x	x	0.069	1.330	1.399
15	x	x	14.3	x	x	x	88	0.336	(0.013)	0.871	0.858
16	x	x	16.6	x	x	x	72	0.348	0.018	0.946	0.964
17	x	x	x	x	x	60	x	x	0.006	1.042	1.048
18	x	x	x	x	x	120	x	x	(0.019)	0.724	0.705

Notes:

Case 1 is base case (Task I preliminary reference design)
x = same as base case
() = negative value
* = wire-wrap spacer; all others use grid-type spacer
Columns 1 through 9 define configuration

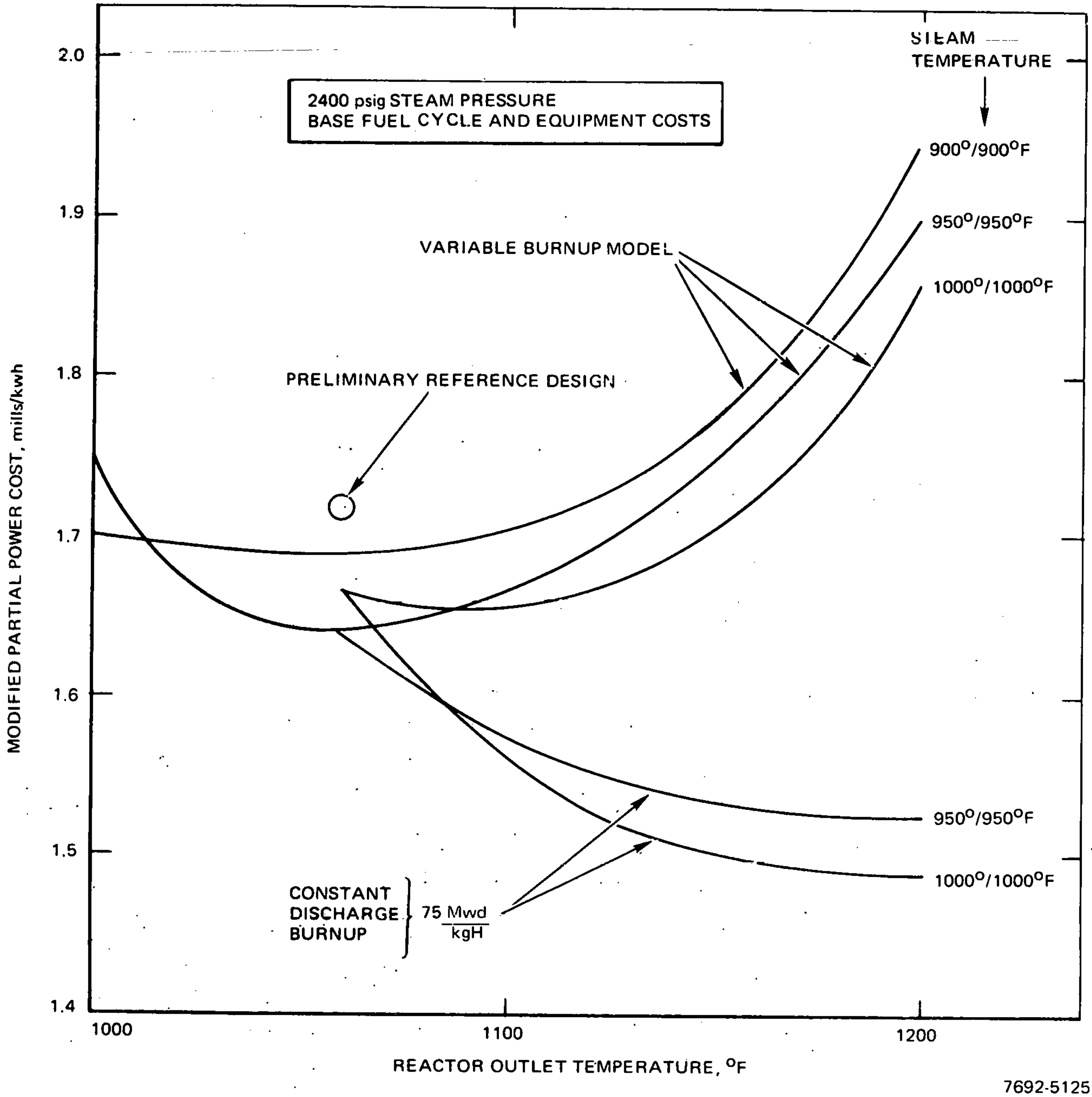


Figure III-1. Partial Power Costs Modified by Temperature Costs vs Reactor Outlet Temperature

4) Optimum system ΔT's, as a function of reactor outlet temperature, were determined.

5) Optimum steam conditions, as a function of reactor outlet temperature, were determined.

B. SELECTION OF REFERENCE DESIGN

The results of the core and system studies indicated several general trends which are:

1) Higher fuel fraction, which can be obtained with larger pins and smaller pin pitches, reduces costs.

2) There is little or no economic incentive for high (>15 kw/ft) linear fuel pin power.

3) The optimum reactor outlet temperature is dependent on the relationship of fuel burnup to reactor outlet temperature, and falls in the range of 1060 to 1200°F.

4) The optimum reactor ΔT, secondary system ΔT, steam generator sodium inlet temperature, and steam conditions are dependent on the reactor outlet temperature selection.

An overall system evaluation, combining the results and further investigating the conclusions of the core and system parameter studies, was made to select the Task III final reference design.

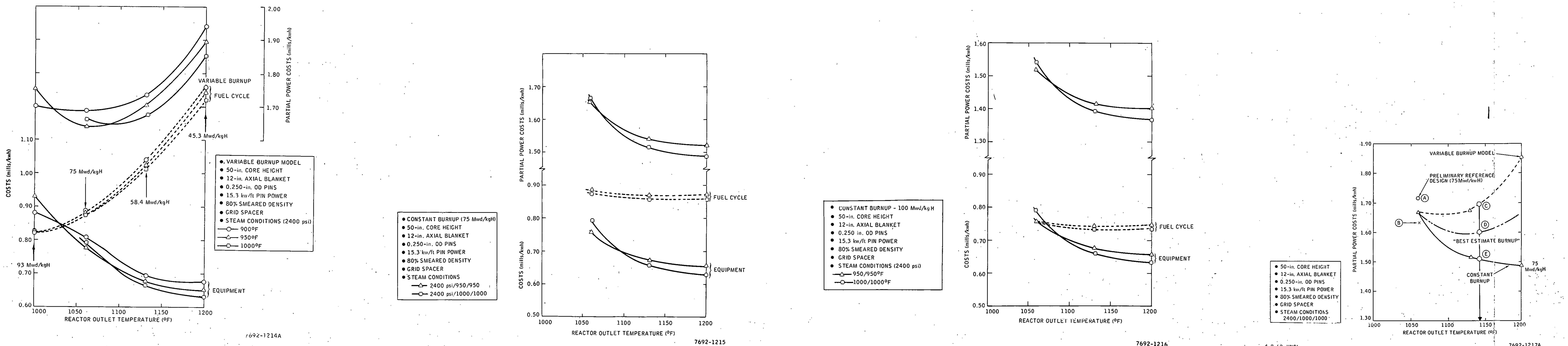
The objective of the Task III design was to provide a conceptual design of an LMFBR which could begin operation in the early 1980's, and be competitive with LWR's. In selecting the design critieria, minimizing the total energy cost was the primary objective, providing the operational date of the early 1980's was not jeopardized, and the technical and safety requirements (Table III-2) were met.

Because of its major effect on the design, the first variable to be examined was reactor outlet temperature. The outlet temperature study was based on variation of the Task I preliminary reference design. Figure III-2a gives the partial power cost contribution of the portion of plant equipment that varies as a function of temperature, and the fuel cycle cost for the variable fuel burnup model. Each point on the curves represents an optimization for the given temperature level, from the standpoint of using the best values of primary and secondary ΔT's. Each curve may be considered as a series of "optimum" design plants. Steam temperatures of 900, 950, and 1000°F, with reheat, were considered. The figure shows that equipment costs decrease as the outlet temperature is increased. This improvement, in general, is due to the decreased heat transfer surface areas required in external equipment, such as the IHX and steam generators, and to the lower plant heat rate. As the temperatures rise, equipment becomes more expensive, because of the greater material thicknesses required as allowable material strength drops off.

The fuel costs, which are based on the variable burnup model, however, substantially increase as the burnup decreases. At 1200°F, the model predicts a burnup level of ~45 Mwd/kgH, compared to 75 Mwd/kgH at 1060°F. The summation of the cost of the portion of the equipment varying as a function of temperature and the fuel cycle cost gives a partial plant power cost value that can be used to compare various designs. The minimum partial power cost for the variable burnup model is at ~1060°F outlet temperature with 950°F steam temperature. Below an outlet temperature of ~1020°F, 900°F steam is best; and, above ~1090°F, 1000°F steam is best.

Figure III-2b shows the same information for a constant burnup of 75 Mwd/kgH. The equipment costs are very similar; but, because of the different optimum system ΔT's which result from the different fuel cycle costs, they are not exactly the same. Fuel cycle costs decrease slightly with temperature. The partial power costs are seen to be very near the minimum at 1200°F outlet temperature. Figure III-2c gives the same information for a constant burnup of 100 Mwd/kgH. The curves have the same characteristics as those in Figure III-2b.

Figure III-2d shows the comparison of results for the different burnup models. For the constant burnup model, an optimum reactor outlet temperature in the order of 1200°F is indicated, and optimum steam conditions for this temperature are 2400 psi/1000/1000°F. Above this optimum temperature, the total costs increase due to (1) higher costs of components, and (2) higher fuel cycle costs, which result from the necessity to increase the wall thickness of the fuel element wrapper tube at higher temperatures. For the variable burnup model, the optimum is at a lower temperature (~1060°F).



a. Variable Burnup Model

b. 75-Mwd/kgH Constant Burnup Model

c. 100-Mwd/kgH Constant Burnup Model

d. Comparison of Burnup Models

Figure III-2. Variation of Costs With Reactor Outlet Temperature

To try to minimize the penalty which might result from selecting the wrong reactor outlet temperature, based on either of the "extreme" models, a model midway between the variable burnup and the constant burnup models was investigated. The "best estimate" burnup model indicates an optimum in the order of 1140°F and this temperature has been chosen for the Task III final reference design.

From Figure III-2d it is seen that, if the "best estimate" burnup model correctly describes the fuel performance, the chosen temperature results in an improvement in power costs of $\sim$0.12 mills/kwh over the preliminary reference design (compare points A and D) and an improvement of $\sim$0.05 mills/kwh over the minimum cost design at 1060°F outlet (compare points B and D). If the variable burnup model describes the performance, the chosen temperature results in a penalty of $\sim$0.04 mills/kwh (compare points B and C). If the constant burnup model describes the performance, the chosen temperature results in an improvement of $\sim$0.14 mills/kwh (compare points B and E). Thus, the "best estimate" burnup model is, in effect, a compromise between risks and potential improvement. The "optimum" steam conditions for a temperature of 1140°F are 2400 psig/1000/1000°F, and the optimum steam generator sodium inlet temperature is 1070°F. Based on this analysis, a reactor outlet temperature of 1140°F, with steam conditions of 2400 psig/1000/1000°F, was selected for the Task III final reference design. This reactor outlet temperature should be achievable by the target opertional date of the early 1980's.

At this higher operating temperature of 1140°F, the "best estimate" burnup model predicts an average discharge burnup in the order of 67 Mwd/kgH, the reduction being due to the assumed loss in ductility of the cladding at higher temperature. The 67 Mwd/kgH was used as the design burnup for the Task III final reference design.

Figure III-3 illustrates the effect of pin pitch on economics for the grid and wire-wrap type spacers considered in the design, and includes all costs which vary as a function of the fuel element pin pitch. These costs include fuel cycle costs, pumping power, and reactor structure capital costs. At constant power, constant pin diameter, and constant reactor ΔT, a decrease in pin pitch increases coolant velocity and reactor pressure drop. This decrease in pin pitch also results in "drier" cores (increased fuel volume fraction), and therefore

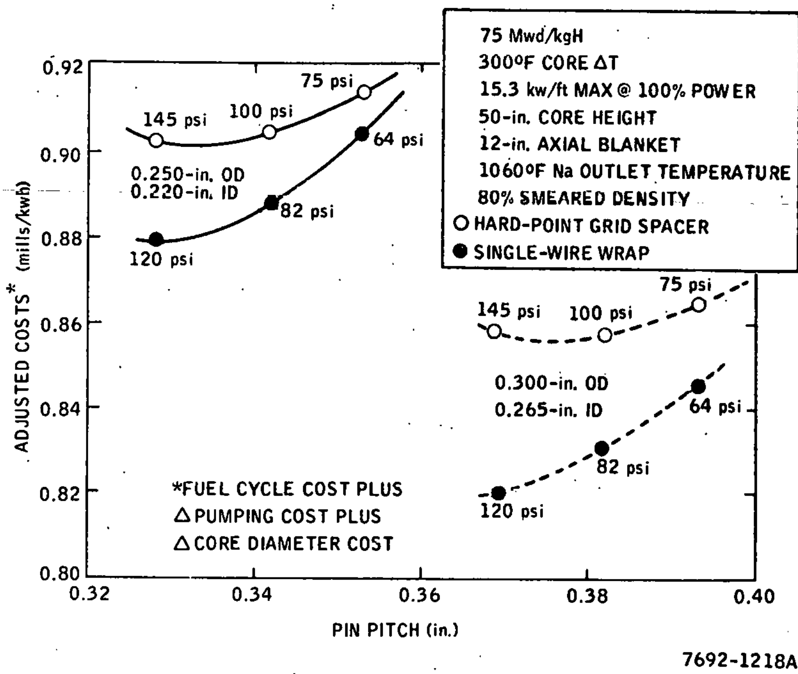


Figure III-3. Pin Pitch vs Adjusted Cost for 0.25- and 0.30-in. Pins

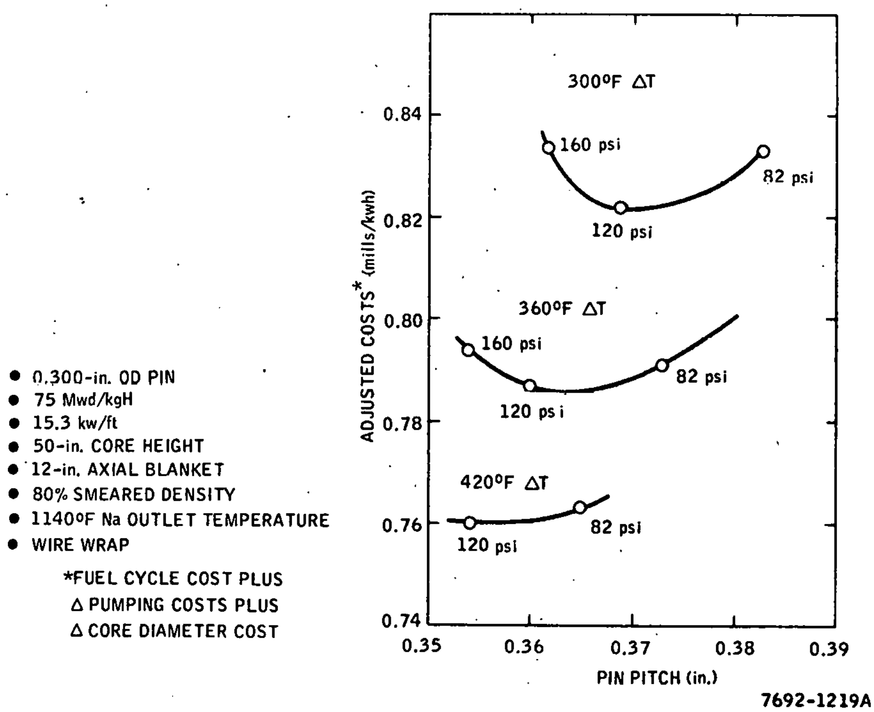


Figure III-4. Pin Pitch vs Costs

tends to lower fuel cycle costs. Figures III-3 and III-4 illustrate that the optimum pressure drop is ~120 psi for a wide variety of design conditions, but costs are relatively insensitive to pressure drop between 82 and 120 psi. The wire wrap is seen to give a significant advantage, because of the drier core which is achieved at the minimum cost point, due to the lower pressure drop characteristic for a given velocity. The wire-wrap spacer also results in an estimated fuel fabrication cost saving of 0.03 to 0.05 mill/kwh, which has not been included. A wire-wrap design has therefore been chosen for the Task III final reference design.

Figure III-3 also illustrates the effect of pin size on overall economics. It is seen that the increase in pin size results in a significant economic improvement, partly from the fact that, at larger pin sizes, the core becomes drier for a given pressure drop, and partly due to the lower fuel fabrication costs resulting from the use of larger pins. This analysis, however was based on a constant radial blanket burnup. Figure III-5 compares the economics of pin size with constant radial blanket burnup and constant radial blanket residence time. It is seen, from this figure, that longer blanket residence times (constant radial blanket burnup) are required to achieve the economic advantage indicated for larger pins. The residence time for the outer radial blanket elements to achieve the estimated optimum exposure in a core using 0.300-in. OD fuel pins is ~7 years. Associated with these long residence times are many potential difficulties, such as fretting, fatigue, corrosion, irradiation-induced distortion, etc. These difficulties have been evaluated only superficially, and the residence time of the blanket elements may be limited by technical factors, rather than economic factors. Although the blanket residence time requires further evaluation to determine the upper limit, we have selected ~7 years as the maximum time for this evaluation. In addition, time-dependent economic factors, which can be properly accounted for only by the detailed fuel-management and cash-flow analyses, become quite important as fuel inventory and residence time increase. The economic and technical factors implicit in the selection of a 0.300-in. OD pin for the reference design need more extensive investigation, which was beyond the scope of our study.

Figure III-6 illustrates how fuel cycle costs, capital costs, and partial energy costs vary with reactor ΔT for the previously selected reactor parameters.

Increasing reactor ΔT dries up the core, thus improving the reactor neutronics (higher breeding ratio and lower fissile mass), and thereby reducing fuel cycle costs in the constant burnup model. In the case of the variable burnup model, higher ΔT's result in higher cladding hot spot temperature, and therefore in lower burnup and consequent higher fuel cycle costs.

The cost of the greater heat transfer areas required at higher ΔT's causes equipment costs to increase. The optimum ΔT for the constant burnup model is ~375°F. For the variable burnup model, the optimum appears to be ~350°F. A compromise of 360°F has been chosen for the reference design as a best estimate with little penality, if the model is wrong in either direction.

Figure III-7 illustrates the effect of core height on plant economics. As the core height is decreased, the fuel element fabrication costs are increased, and the number of fuel elements required (for constant element size) is increased, leading to increased fuel handling time. The increased fuel handling time produces an unavailability penalty, as core height is decreased. The unavailability penalty was included only if the total reactor fuel handling time per year exceeded that required for an annual turbine maintenance schedule (assumed to be 2 weeks/year). Because the coolant flow per element becomes less and the sodium volume fraction is reduced, in order to hold the optimum pressure drop, a decrease in fuel cycle cost is obtained as core height is decreased. Shorter cores also reduce capital costs for the selected system configuration. These competing effects produce an optimum in core height, dependent upon which burnup model is assumed. If the constant burnup model is assumed, the optimum core height is ~42 in. If the variable burnup model is used, the optimum is ~38 in. This difference is due to the fact that the axial peak-to-average power ratio becomes less, as the core height is decreased; and, for the variable burnup model, this results in a higher fuel burnup. Because of the flatness of this curve, the core height can be adjusted over a range of ~38 to 43 in. without significantly influencing the economics. Therefore, this parameter, in addition to the linear power rating of the fuel, is used to adjust the core design to achieve a specific power consistent with the 1-year refueling period and the required integral number of fuel elements, to give a good hexagonal pattern for the core layout. This results in a tentative core height selection of 43 in. for the final reference design.

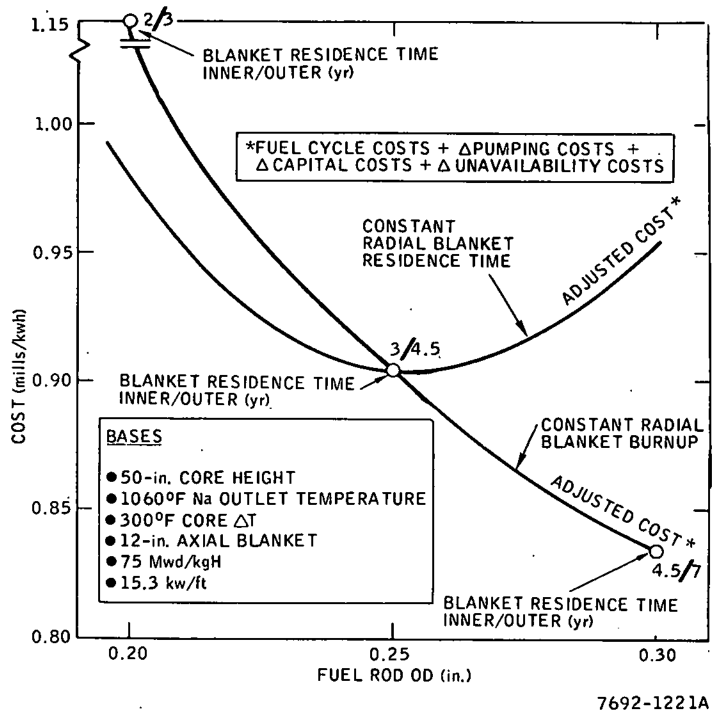


Figure III-5. Effect of Radial Blanket on Economics of Pin Diameter

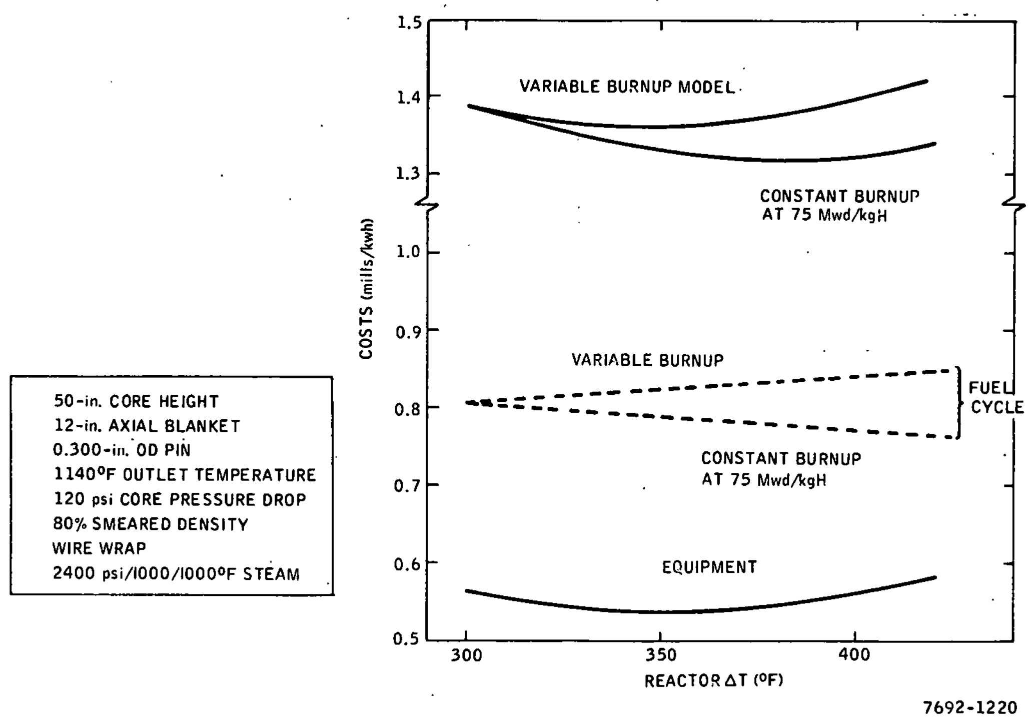


Figure III-6. Partial Power Costs vs Reactor ΔT for 0.30-in. Pin

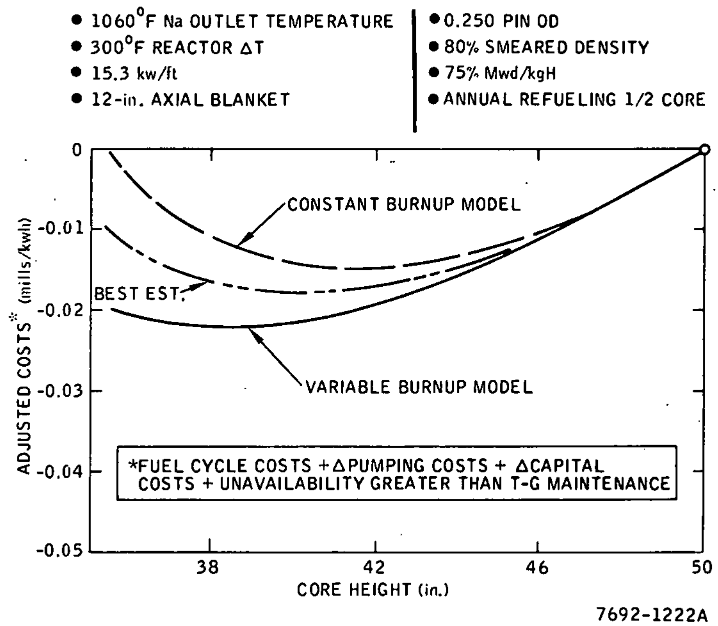


Figure III-7. Cost Variation With Core Height

C. SUMMARY OF SELECTION

On the basis of analyses and evaluation of the parameter studies and the design studies, a Task III final reference design was selected. The selection of some of the design options and independent variables, relating to the plant systems and reactor building arrangement, are summarized later in the report. To make the definition of the final reference design presented in this section complete, the following items are included.

1) Steam generator module size

2) Number of heat transfer circuits

3) Number of IHX units per heat transfer circuit

4) Spent fuel storage facilities

5) Reactor building size.

Table III-5 lists the lists the major changes from the Task I preliminary reference design, and summarizes the approximate economic impact of each of the selected design and/or performance improvements incorporated in the final reference design. The total reduction in energy cost due to each improvement is the net result of the changes in capital, fuel cycle, and availability costs for the particular improvement. These improvements have been further grouped into two categories – those that apply at the 1060°F reactor outlet temperature (Lines 1 through 10), and those that result from the increase to 1140°F outlet temperature (Line 12). The combined worth of these improvements is 0.52 to 0.67 mill/kwh, corresponding to a burnup range of 67 to 100 Mwd/kgH. Note that ~75% of this cost improvement may be realized without the use of higher reactor outlet temperature; and a plant with 1060°F reactor outlet temperature is, in fact, competitive under the definition discussed earlier. This is quite significant, since the increased outlet temperature design will require considerably more R&D, particularly in in-pile fuel cladding behavior at ~1300°F. We therefore conclude that an LMFBR of the basic type studied by AI will provide the 10% cost incentive estimated to be needed to assure the wide-scale application of LMFBR's by utilities without increasing the reactor outlet temperature above 1060°F. Increasing the reactor outlet temperature to 1140°F, however, will take further advantage of the LMFBR's potential for providing

TABLE III-5

APPROXIMATION OF ECONOMIC EFFECTS OF IMPROVEMENTS TO THE TASK I PRELIMINARY REFERENCE DESIGN

PARAMETER	CHANGE IN PARAMETER	TOTAL ENERGY COST REDUCTION (mills/kwh) (a) + (b) + (c)	CAPITAL COST* 10^3 ($)	CAPITAL COST* (mills/kwh) (a)	FUEL CYCLE COST* (mills/kwh) (b)	AVAILABILITY* (%)	AVAILABILITY* (mills/kwh) (c)
1. REACTOR BUILDING DIAMETER AND NUMBER OF IHX's/LOOP	176 — 140 ft, 2 — 1	0.046	2,450	0.046	-	-	-
2. STEAM GENERATOR MODULES	91 — 380 TUBE	0.097	5,200	0.097	-	-	-
3. AUSTENITIC STEAM GENERATOR		0.014	(350)	(0.006)	0.020	-	-
T_{SI}**	950 — 1000°F						
T_{STEAM}	900/900 — 950/950°F						
4. TURBINE TYPE	X-COMPOUND — T-COMPOUND – 3600/1800 rpm 3600 rpm	0.017	1,820	0.034	(0.017)	-	-
5. REFUELING PERIOD	6 — 12 months	0.041	-	-	-	2.0	0.041
6. PIN SIZE (in.)	0.250 — 0.300	0.047	(480)	(0.009)	0.056	-	-
7. SPACER TYPE	GRID — WIRE WRAP	0.077	(110)	(0.002)	0.079	-	-
8. CORE HEIGHT (in.)	50 — 43	0.016	480	0.009	0.008	(0.05)	(0.001)
9. SMEARED DENSITY (% T.D.)	80 — 85	0.029	-	-	0.029	-	-
10. LOWER AXIAL BLANKET THICKNESS (in.)	12 — 15	0.004	-	-	0.004	-	-
SUBTOTAL WITHOUT INCREASE IN OUTLET TEMPERATURE AT 75 Mwd/kgH [SUM ITEMS (1) THROUGH (10)]		0.388	9,010	0.169	0.179	2.05	0.040
11. FUEL BURNUP (Mwd/kgH)	75 — 100	0.086	-	-	0.086	-	-
SUBTOTAL WITHOUT INCREASE IN OUTLET TEMPERATURE AT 100 Mwd/kgH [SUM ITEMS (1) THROUGH (11)]		0.474	9,010	0.169	0.265	2.05	0.040
12. REACTOR OUTLET TEMPERATURE (T_{RO}) WITH AUSTENITIC STEAM GENERATOR							
T_{RO}	1060 — 1140°F	0.134	8,300	0.154	(0.020)	-	-
T_{SI}	1000 — 1070°F						
T_{STEAM}	950/950 — 1000/1000°F						
FUEL BURNUP (Mwd/kgH)	75 — 67						
SUBTOTAL AT 1140°F REACTOR OUTLET TEMPERATURE AND 67 Mwd/kgH [SUM ITEMS (1) THROUGH (10) AND (12)]		0.522	17,310	0.323	0.159	2.05	0.040
13. FUEL BURNUP (Mwd/kgH)	67 — 100	0.150	-	-	0.150	-	-
TOTAL ALL IMPROVEMENTS [SUM ITEMS (1) THROUGH (10), (12), AND (13)]		0.672	17,310	0.323	0.309	2.05	0.040

*FIGURES IN PARENTHESES ARE COST PENALTIES
**T_{SI} = SUPERHEATER SODIUM INLET TEMPERATURE

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electrical energy at costs significantly below other sources, and represents a reasonable goal for the 1980's. The 1140°F outlet temperature has been selected for the Task III final reference design for these reasons, recognizing that, even if the required R&D in support of the higher outlet temperature design could not be completed, the LMFBR would still be competitive on the selected time scale.

D. PLANT ARRANGEMENT STUDIES

The plant arrangement developed for the preliminary reference design in the Task I portion of the work was studied in more detail in Task III. The purpose of this effort was to develop the plant arrangement design in more detail, in order to permit a more accurate capital cost estimate, and to further investigate areas of the design where capital costs could be appreciably reduced. The resulting plant arrangement which was selected for the final reference design represents one which could be built in the 1980's. The design is based on reasonable extrapolation of present day technology, and its construction is assumed to follow the construction and operation of large (300 to 500 Mwe) demonstration plants in the late 1970's.

Work on the plant arrangement was concentrated exclusively on the nuclear island, with primary emphasis on the reactor building design. Preliminary cost estimates, done under the Empire State Atomic Development Associates (ESADA) 1000-Mwe FBR Program for a reactor building similar to that developed for the preliminary reference design (176-ft diameter building), indicated that the building structural and mechanical costs were in excess of $\$8 \times 10^6$. Rough estimates of costs for similar structures result in order-of-magnitude costs of $40,000 per foot of diameter. With this in mind, it appeared that the reactor building design represented the area within the plant arrangement effort in which the most significant cost reduction could be achieved. Consequently, the principal effort involved an attempt to reduce the size of the reactor building and simplify its internal arrangement.

One of the first alternatives considered, in an effort to reduce plant costs, was the use of a single IHX per heat transfer circuit, rather than the two units per circuit used previously. A cost comparison was made to determine the incentives involved in the use of single units.

The approximate cost incentive for using a single unit, rather than the dual units, is $500,000. Inclusion of savings in the cost of the IHX units will further increase the incentive. Also, the cost penalty (for a single unit) for an increase capacity could be eliminated, as was ultimately done for the final reference design, on the basis that IHX removal is not expected to be frequent enough to justify the larger crane size. From a standpoint of fabrication and shipping, the units, even for a two-loop plant with single units per loop, are within present day capability.

The IHX used in the 1000-Mwe plant is a single-pass, fixed-tube-sheet unit of a "hockey-stick" configuration. The unit has a 90° bend in the shell, designed to provide for differential thermal growth between shell and tubes.

Several concepts were considered, prior to the selection of the "hockey-stick" unit. The work was done under the AI-funded Demonstration Plant Program. The program included limited consideration of IHX's for a 1000-Mwe plant. The following concepts were evaluated in the study:

1) Hockey-stick - fixed tube sheet
2) Sine wave tubes - fixed tube sheet
3) Removable tube bundle - floating-head straight tubes
4) Straight tubes - fixed tube sheet - expansion joint in shell
5) Horizontal U-tube - fixed tube sheets
6) Straight tube - floating head - pipe "expansion joint" (double-wall pipe).

The most economical of the concepts is the straight tube unit with an expansion joint in the shell, similar to that used at the Hallam Nuclear Power Facility. However, the present nuclear vessel code prohibits expansion joints in Class A vessels; and, even with the secondary sodium on the shell side, it is possible that this concept would not receive code approval. Consequently, this concept was not chosen. The hockey-stick concept was selected on the basis that, with the exception mentioned, it is the most economical design.

Each of the three primary heat transfer loops in the preliminary reference design contains two block valves, used for loop isolation. The valves are not required for safety purposes. They are included to isolate the loops for maintenance purposes. The valves represent a significant capital cost item, and consideration was given to the possibility of eliminating them from the design.

Considering a design without valves, if a single primary loop requires shutdown and vault access, the entire plant has to be shut down while the total primary system decays. When vault access is permitted, the adjacent loops have decayed; consequently, heavy shielding between loops is not required. Only atmospheric separation would likely be required, to permit maintaining an inert atmosphere over the loops still circulating to remove decay heat.

Deletion of the block valves from the design represents a capital cost savings in the order of $820,000. The principal advantage of having the valves is an increase in overall plant availability during single-loop shutdown. Being able to isolate a single loop for decay, while operating the plant at two-thirds power, saves ~$320,000, because of the improved plant availability.

The general conclusion of the investigation was that it is difficult to justify the elimination of block valves on a purely economic basis. Deletion of the valves can only be justified on the basis of long term, successful, operating experience indicating that they are unnecessary.

Several ways of handling and storing fuel external to the reactor vessel were investigated. The concepts investigated included handling and storage facilities both inside and outside the reactor building. The final selection was storage in sodium in the reactor building for the normal fuel change. Spent fuel shipping cask loading is inside the building. This selection was based on economics and safety. From a safety standpoint, having all storage and cask loading inside the reactor building is the most desirable approach. With this system, all spent fuel is doubly contained (cask and seal-welded canister) before leaving the building, hence the possibility of radioactive release to the atmosphere is reduced to a minimum. From an economic standpoint, the elimination of any storage in excess of that required for normal fuel change-over is worth in the order of $500,000 in storage facilities alone.

The plant arrangement studies resulted in substantial changes over the Task I preliminary reference design. The selections made for the final plant arrangement and the basis are summarized in the following sections.

1. Number of IHX's

Selection — Single IHX unit per heat transfer circuit.

Basis — The single-unit arrangement permits a substantial reduction in building size, and a corresponding reduction in plant capital cost.

2. IHX Configuration

Selection — Shell-and-tube hockey-stick IHX configuration.

Basis — The straight shell and tube IHX, with a bellows in the shell for thermal expansion, is the minimum cost concept. This use of a bellows in the shell is not acceptable under the present ASME code for Class A nuclear vessels. The hockey-stick configuration was selected, as it is the next lowest in cost.

3. Primary Loop Isolation Valves

Selection — Block valves will be used for primary loop isolation.

Basis — Block valves permit isolation of one loop of a three-loop plant, while continuing operation on the two remaining loops. This capability allows a loop requiring unscheduled maintenance to be shut down and the sodium activity to decay while the plant operates at about two-thirds power.

4. Ex-Vessel Fuel Handling Arrangement

Selection — Ex-vessel storage in sodium, for a normal refueling batch, inside the reactor building. Preparation of spent fuel for shipment, and shipping cask loading, inside the building.

Basis — This is the lowest cost arrangement of those studied, and meets all identified requirements.

E. STEAM GENERATOR TRADE STUDY

The Steam Generator Trade Study compared the costs of a modular steam generator installation, relative to a large unit steam generator installation, for a 1000-Mwe LMFBR plant. Each steam generator installation consists of evaporators, superheaters, and reheaters. In the modular concept, each function is accomplished in separate multiple heat exchangers for each loop. In the single large steam generator concept, as typified by the Babcock-Wilcox Company design, the evaporator and superheater are unitized into one large heat exchanger in each loop, with a separate reheater. With a module concept, unit size may be increased until only a single module is used for each function (evaporation, superheat, and reheat).

The comparison of the two steam generator concepts was on the basis of the AI Task I preliminary reference design system conditions.

Since the basic objective of the trade study was to investigate the variation of costs as the sizes of the steam generators were varied, the size of modular units, up to a single evaporator, superheater, and reheater per loop, was investigated. Five arrangements were considered, varying from 36 evaporators, 16 superheaters, and 8 reheaters per loop to 1 evaporator, 1 superheater, and 1 reheater per loop. The cost comparison included first (capital) costs, unavailability (loss of power) costs, and maintenance (repair or replacement due to a leak) costs. Each of these factors was considered on the basis of using, or not using, spares.

The total incremental levelized cost for the modular steam generators as a function of size were determined. Both with and without spare modules, the use of multiple modules results in an appreciable cost savings, related to the use of single modules. The indicated cost savings, using the minimum-cost module size, are $\sim$0.110 mills/kwh without spares, and 0.130 mills/kwh with spares, which is equivalent to $\sim\$6.0 \times 10^6$ to $\$7.0 \times 10^6$ in capitalized cost.

The optimum module size appears to be in the vicinity of 400 to 600 tubes per module. For the small modules, as the size is increased, the fabrication costs and related equipment costs decrease at a faster rate than the unavailability and maintenance costs increase; however, in the larger sizes, the related equipment costs decrease slightly, and the fabrication, unavailability, and maintenance costs increase.

F. NUCLEAR UNCERTAINTY ANALYSIS

There are large uncertainties in many of the nuclear cross sections needed for LMFBR analysis and design. These uncertainties can result in substantial differences between predicted and "as built" characteristics. An analysis has been performed to determine the uncertainties that should be assigned to the critical mass and breeding ratio, and to identify additional work to reduce them. The effect of these uncertainties on fuel-cycle costs has been estimated.

The method used to evaluate the accuracy of the cross-section set used in the 1000-Mwe LMFBR Follow-On Studies was to correlate analytical results, using several cross-section sets, with critical assembly experimental data. This correlation was made with five large plutonium-fueled critical experiments, using the basic AI-FBR cross-section set and two separate modifications to this set. The same cross-section variations and the same analytical methods tested in the experimental correlations were also used to analyze the Task I preliminary reference reactor design, to determine the influence of cross-section uncertainties on breeding ratio and fissile mass. This procedure provided an accurate assessment of the uncertainties to be assigned to the breeding ratio and assembly reactivity of the 1000-Mwe LMFBR. This work also provided guidance for the selection of an improved cross-section set to be used for the final Task III design analysis.

The major conclusions drawn from this work were:

1) The effective multiplication constant (k_{eff}) of the Task I preliminary reference design is overpredicted by 1-1/2 to 3-1/2% Δk. (This produces a fissile mass underprediction of 2.7 to 6.3%.

2) The Gwin-Dunford-ENDF/B Data library (higher Pu^{239} α, lower U^{239} capture cross sections) appear to be better than the Basic AI Data used for Task I work and for the Plant Design Studies. These changes will reduce the breeding ratio by 0.08.

3) The integral α experiments cannot be used to test the validity of Pu^{239} α values being used in a cross-section library.

4) The fuel-cycle cost uncertainty, due to cross-section variations which should be assigned to the Task I preliminary reference design, is +0.159, -0.09 mills/kwh. This uncertainty applies to calculations which have been made, using the Basic AI Data.

Other conclusions are:

1) The ZEBRA 7 assembly data is the most representative of the Task I preliminary reference design, having a spectrum that is quite close to that calculated for the reactor.

2) There is a 2 to 3% Δk overprediction of reactivity with either the Basic AI or the Gwin-Dunford-ENDF/B Data. Calculations should be performed on the five critical assemblies with the σ_f of Pu^{239} decreased, when the Gwin α values are used (only σ_c was changed in the correlations). The degree of uncertainty in ν should be checked to see whether a lower value is desirable. Also, the influence of the possible variations in χ (E) should be checked, using these critical assemblies. The other items which can cause significant errors in the eigenvalue correlation are the heterogeneity and transport corrections. The heterogeneity effect in the correlations are ~2% $\Delta k/k$, and therefore of the same magnitude as the overprediction in k_{eff}.

3) The effect of Pu^{239} α values which are higher than the Gwin data (nearer to Schomberg and Sowerby's data) should be tested on the correlations, to see whether this would be better than the Gwin-Dunford-ENDF/B Data for correlating both the central reactivity coefficients and the effective multiplication constants (k_{eff}) of the criticals.

G. COMPARISON OF REGULAR AND MODULAR CORE GEOMETRIES

The effort on the comparison of a regular geometry and a modular core is limited to a direct comparison of prior work. (The term "regular geometry" is used to mean a core design in which height and diameter are chosen, based on economic considerations along.) The Task I preliminary reference design was compared with a modular core design prepared on an earlier AEC study. The major features, characteristics, and analytical methods for the two designs were comparable. The fuel material (oxide), fuel element configuration, peak nominal linear fuel pin power, and plant operating conditions were the same for both designs. The same basic nuclear cross-section set was used for both analyses. The core thermal-hydraulic analysis was performed, using the FAITH computer code described in Volume V.

The equilibrium fuel cycle costs for both cores are given in Table III-6. Comparison of these costs shows that there is about a 0.15 mill/kwh penalty for a modular core, as compared to a regular geometry core. This compares

TABLE III-6
EQUILIBRIUM FUEL CYCLE COSTS

	Regular Geometry Core (mills/kwh)	Modular Core (mills/kwh)
Fabrication	0.470	0.482
Fabrication Working Capital	0.041	0.045
Reprocessing	0.176	0.203
Shipping	0.049	0.056
Reprocessing and Shipping Working Capital	-0.031	-0.038
Plutonium Sales	-0.295	-0.296
Plutonium Inventory	0.444	0.555
Total	0.854	1.007

well with a previous study done by AI for a 500-Mwe plant size,* which showed an 0.13 mill/kwh penalty for a modular core. The referenced study also showed that, of the spoiled core geometries, the modular core suffers the least economic penalty.

H. SAFETY STUDIES

Most reactor accidents may be grouped into two classes:

1) Those which may be expected to occur on the order of once in the lifetime of the plant (Class I)

2) Those which are highly unlikely to occur during the plant lifetime (Class II), but still can be postulated.

Examples of Class I accidents are loss of pumping power, and small gas bubbles. Examples of Class II accidents are a guillotine pipe rupture, a large gas bubble, and the handling, in the ex-core fuel handling machine, of a fuel element which has been removed prematurely and has a high decay power level.

*H. M. Dieckamp et al., "Atomics International Demonstration Fast Breeder Reactor," ANS Fast Reactor National Topical Meeting, San Francisco, Calif., April 1967

Figure III-8 illustrates the safeguard actions required in the plant design for the two classes of accidents. It should be noted that unprotected Class I and Class II accidents can lead to comparable consequences. However, because of the assumed greater probability of Class I accidents, two independent safeguards are provided, followed by containment should both safeguards fail. Because of the very low occurrence probability of Class II accidents, only one level of safeguards is provided prior to requiring containment.

Consideration of this discussion indicates that it may be possible to design a highly reliable protective system, with greater confidence than an equally reliable containment system for core disassembly accidents. Those accidents which do not involve the core, but release radioactive material to the containment system (such as a large sodium fire), can be contained with a very high level of confidence. Then, if an additional safeguard were added for all Class I and Class II accidents which involve the core, a safer plant might result, even with the removal of features which presently are provided to contain a core disassembly accident. This is the alternate approach shown on Figure II-8. Such an approach could lead to a more economic plant with equal safety. This should be pursued in future studies, but was not studied in the LMFBR Follow-On Study.

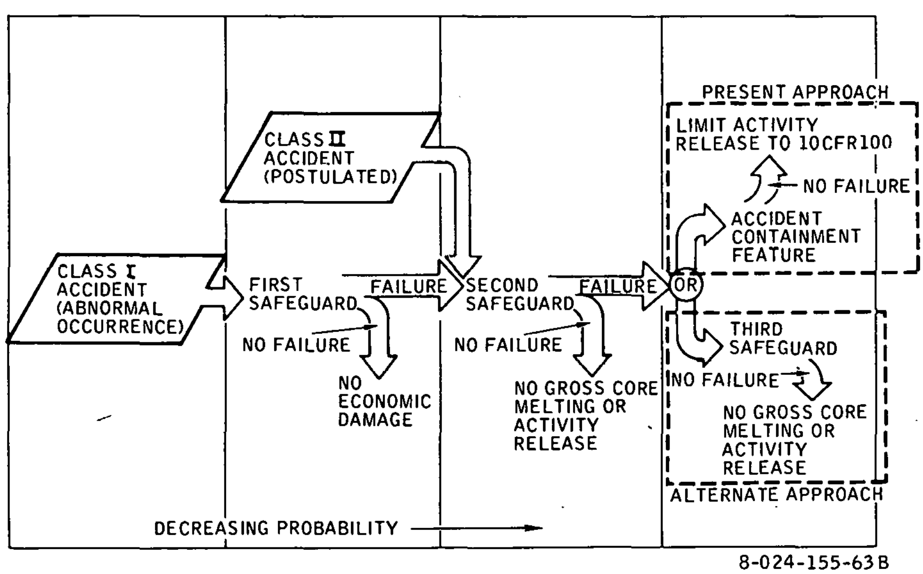

Figure III-8. Accident Sequence and Safeguards

The approach, in this study, is one of providing containment for a core disassembly accident. This approach was selected to demonstrate LMFBR plant safety capability; becuase it is based on precedent, and is believed to be conservative, with respect to showing that a 1000-Mwe LMFBR is economically viable. Furthermore, it was uncertain as to whether, at this time, the additional safety features required for the alternate approach could be defined well enough to assess their performance. It was therefore concluded that the objectives of the program could best be met by:

1) Designing the reference plant, based on containment of a disassembly accident

2) Recommending a development program which would allow a decrease in the large design margins for present containment uncertainties

3) Recommending additional development which would lead to the inclusion of additional safeguards and protective features that would preclude any reasonable probability of a core disassembly accident.

A review and screening of the potential accidents associated with the Task I preliminary reference design resulted in the following accidents being judged as having the potential for significantly influencing the design:

1) Unprotected* loss of coolant flow to the reactor

2) Unprotected* guillotine pipe rupture with double-ended flow

3) Large gas bubble (ρ > $1.50) passing through the core of the reactor

4) Large sodium spill in an open primary system vault

5) Overheating of fuel elements in the ex-core fuel handling machine.

The unprotected loss of coolant flow was selected as the Design Basis Accident (DBA) which sets the building leak rate; but, as will be seen, several of these accidents will have an influence on the plant design.

To estimate the credibility and design implications of these accidents, fault trees were constructed, and detailed analyses of the events following the accident initiation were performed. Based on these studies, design or procedural

*Protective system fails to operate

requirements were established to minimize the probability of initiating the accident, and the required protective action was identified, as shown in Table III-7.

It was concluded that the design measures listed in Table III-7 would adequately preclude the possibilities of a rod ejection accident. It was also concluded that a loss-of-site-power accident, with subsequent protected loss of reactor flow, was a Class I accident. Qualitative judgements were made of the probability of the occurrence of a given accident, relative to the protected loss-of-coolant-flow accident (loss of pump power). From these judgments, shown in Table III-7, AI deemed that the pipe rupture, large gas bubble, open vault sodium spill, and loss of cooling in the fuel handling machine accidents are Class II accidents.

The last column in Table III-7 (the product of the preceding two columns) represents the judged probability of occurrence of an unprotected accident, relative to a loss-of-flow accident. From the evaluation, it can be concluded that five of the accidents need to be considered for containment. The detailed evaluation of the unprotected accidents results in approximately equivalent consequences for the two most severe accidents: (1) the unprotected loss of flow, and (2) the unprotected guillotine pipe rupture. Because the loss-of-flow accident is the most probable, it was selected as the DBA. Therefore, the DBA consists of the following postulated sequence of events:

1) Power is lost to the pumps.

2) Control system action fails.

3) Both the primary and secondary protective systems fail to act, and the control rods fail to drop or be driven in.

4) No effective operator action is taken.

5) Voiding of the core is initiated in 7 to 10 sec, as the flow decreases to $\sim$25% of its initial value.

6) Excessive reactivity is inserted, following core voiding, due to the positive core void coefficient. The maximum reactivity insertion rate is 66 \$/sec, when \$1.50 has been inserted.

TABLE III-7

QUALITATIVE ACCIDENT SELECTION

ACCIDENTS	DESIGN OR PROCEDURAL ACTION	PROTECTIVE ACTION	ESTIMATED PROBABILITY OF ACCIDENT INITIATION RELATIVE TO LOSS OF FLOW ACCIDENT	ESTIMATED PROBABILITY OF NO PROTECTIVE ACTION RELATIVE TO LOSS OF FLOW ACCIDENT	ESTIMATED PROBABILITY OF AN UNPROTECTED ACCIDENT RELATIVE TO THE UNPROTECTED LOSS OF FLOW ACCIDENT
CORE EXCURSION ACCIDENTS					
LOSS OF COOLANT FLOW	1) ALTERNATE SOURCES OF PUMP POWER 2) DESIGN HIGHLY RELIABLE PUMP AND PUMP DRIVE SYSTEMS	REACTOR TRIP ON HIGH POWER-TO-FLOW RATIO WITH HIGH TEMPERATURE AS A BACKUP SIGNAL	1 (REFERENCE)	1 (REFERENCE)	1 (REFERENCE)
PIPE RUPTURE	1) ASSURE DUCTILE PIPING DURING ALL OPERATING CONDITIONS 2) PROVIDE ADEQUATE STRESS DESIGN MARGINS 3) PROVIDE SODIUM LEAK DETECTION OF SMALL LEAKS	REACTOR AND PUMP TRIP ON LOSS OF SODIUM LEVEL IN REACTOR VESSEL	$<<1$ (MUCH LESS)	$>>1$* (SIGNIFICANTLY GREATER)	~ 1 (COMPARABLE)
LARGE GAS BUBBLE	1) DESIGN TO MINIMIZE ENTRAINMENT AND SOLUBILITY 2) DESIGN TO AVOID ALL POSSIBLE GAS COLLECTION SITES 3) PROVIDE VENTS AT ALL REMAINING POTENTIAL GAS COLLECTION SITES 4) DESIGN SYSTEM TO BREAK UP AND DISPERSE ALL LARGE GAS BUBBLES	REACTOR TRIP ON HIGH POWER-TO-LOW-FLOW SIGNAL FROM GAS BOUND FLOWMETER	$<<1$ (MUCH LESS)	$>>1$ (SIGNIFICANTLY GREATER)	~ 1 (COMPARABLE)
ROD EJECTION	1) DESIGN TO MINIMIZE PROBABILITY OF SUBJECTING ROD TO SUFFICIENT PRESSURE TO EJECT IT IF NOT HELD DOWN 2) DESIGN TO PROVIDE ADEQUATE HOLDING FORCE TO PREVENT EJECT ON EVEN IF MAXIMUM THEORETICAL PRESSURE FORCE WERE AVAILABLE 3) DESIGN TO AVOID MELTING OF ROD CLADDING IN EVENT OF BLOCKAGE OF COOLANT INLET	NONE	~ 0 (NEGLIGIBLE)	$>>1$ (SIGNIFICANTLY GREATER)	~ 0 (NEGLIGIBLE)
EX-CORE ACCIDENTS					
LARGE SODIUM SPILL	1) PROVIDE ADEQUATE PROCEDURES TO AVOID GROSS ERROR RESULTING IN A LARGE SODIUM SPILL WITH VAULT OPEN	1) VALVE INTERLOCKS 2) ADMINISTRATIVE CONTROL	$<<1$ (MUCH LESS)	$>>1$ (SIGNIFICANTLY GREATER)	~ 1 (COMPARABLE)
LOSS OF COOLING IN THE FUEL HANDLING MACHINE	1) PROVIDE REDUNDANT COOLING CAPABILITY 2) ESTABLISH PROCEDURES AND FUEL HANDLING MACHINE PROGRAMMING TO PREVENT REMOVAL OF A HIGH POWERED ELEMENT 3) PROVIDE LOW LEAKAGE SEALS TO CONTAIN RELEASED FISSION GASES WITHIN FUEL HANDLING MACHINE	1) PROVIDE INSTRUMENTATION TO DETECT LOSS OF COOLING 2) REMOTE MEANS OF RETURNING FUEL TO REACTOR STORAGE POSITION	$<<1$ (MUCH LESS)	$>>1$ (SIGNIFICANTLY GREATER)	~ 1 (COMPARABLE)

*ACCORDING TO THE REFERENCE APPROACH ONLY ONE SAFEGUARD SYSTEM IS PROVIDED FOR CLASS II ACCIDENTS.

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7) The transient is terminated by disassembly of the core.

8) The fuel mixes and comes to thermal equilibrium with the sodium in the core, and the sodium vapor subsequently insentropically expands, resulting, in an estimated 2200 Mw-sec of work.

9) The top shield rises, releasing sodium and fission products to the containment dome.

10) Fission products, plutonium, and sodium are vented to the reactor cavity and to the heat transfer vaults through rupture discs. They are also vented from the reactor vessel, if it ruptures.

11) Some fission products, plutonium, and sodium leak from the inner containment to the outer containment.

12) A small fraction of these radioactive products leak through the outer containment structure to the outside.

Evaluation of the protection requirements of the pipe rupture or vessel leakage accidents resulted in the selection of elevated heat transfer loops. The three elevated loops provide redundant emergency free-convection cooling. The elevated loops, the in-vessel inlet lines, and the pump and IHX pits prevent siphoning of the sodium below the safe level for convection flow in the remaining loops. A jacket is provided for the vessel, to assure maintenance of a safe sodium level, in the event of a vessel leak.

Evaluation of the DBA resulted in the plant containment features shown in Figure III-8. Following the postulated core excursion, the top shield motion is controlled by the top shield holddown structure. This structure subsequently transmits the load to the building. Inert gas (2% oxygen maximum) is provided in the inner containment, which is designed for a maximum leakage rate of 10%/day at 20 psig and 100°F. To provide attenuation of the fission product release, an outer containment is provided which has a design leakage rate of 0.5%/day at 10 psig and 100°F. Additional features required are the building air filter, building concrete shielding, missile barrier for the control rod drive system, and the emergency cavity liner cooling system.

The alternate approach (indicated on Figure III-8), of providing additional protective features and safeguards to eliminate the core excursion as the DBA, is expected to have the net effect of increasing the overall plant safety and reducing the capital investment. This would result in the large sodium fire becoming the DBA, and would allow the following changes in the containment features shown in Figure III-9:

1) Removal of the containment dome and its storage structure
2) Elimination of the top shield holddown
3) Considerable decrease in the size of the support ring structure
4) Considerable decrease in structural requirements of the inner containment
5) Removal of test channels from weld seams in inner containment liner, and reduction in quantity of liner plate
6) Considerable decrease in the air cleanup system
7) Removal of the reactor cavity emergency cooling system.

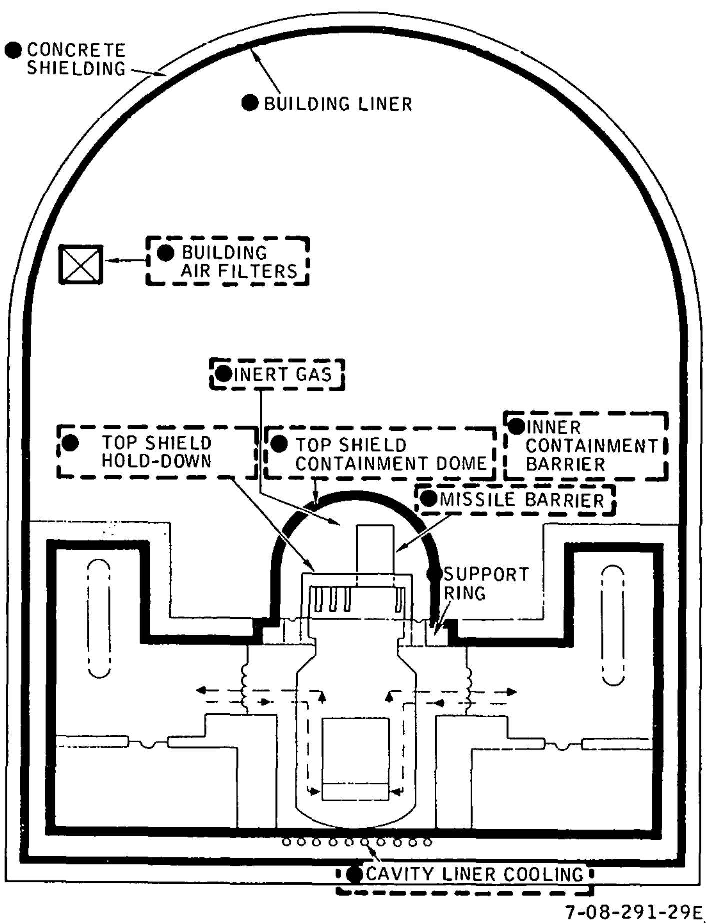

Figure III-9. Plant Containment Features

The capital cost savings which would result from these changes have not been estimated. In addition, the absence of the containment dome and top shield holddown would result in some savings in refueling time. Thus, this alternate approach would likely provide increased safety at lower cost.

The maximum pressure loads for the inner containment are set, by the DBA, to be 51, 22, and 20 psig for the reactor cavity, containment dome, and equipment vaults, respectively.

The maximum pressure loads for the building are set, by the open-cell sodium spill and subsequent fire, to be 2.2 psig. This pressure is reached in 1.9 hr after the accident, and the maximum building gas temperature is $\sim$180°F. The maximum building pressure reached during the DBA is 1.2 psig.

A brief summary of the results of analysis of other accidents investigated for the Task I preliminary reference design is as follows:

In the ex-core fuel handling machine accident, both normal and emergency cooling systems (each with 60-kw capacity) were assumed to fail, with failure of the automatic fuel removal programming resulting in a 120-kw fuel element being placed in the fuel handling machine without adequate cooling. After failure of a number of safeguards, the fuel element cladding is assumed to overheat and fail, releasing the fission product inventory to the fuel handling machine.

A shim-safety control rod ejection accident was analyzed, based on gross multiple failure of the lower core structure, thus placing high-pressure sodium under the poison bundle of a control rod with follower withdrawn from the core. Ejection could result in a reactivity insertion rate of 6 \$/sec which, if unprotected, results in a power excursion of lesser consequence than the DBA loss-of-flow accident. However, after considering the various types of core-structure failure required to give sufficient pressure to lift the poison bundle at a significant velocity, and the need for a control rod to be delatched at the time, it is concluded that the posed structural failure is of sufficiently low probability of occurrence to be deemed incredible.

A plugged coolant channel accident was examined for partial plugging and complete element blockage. The former condition would lead to overheating of the cladding (>1300°F), if the length of the plug is greater than one fuel rod diameter, for one triangular flow passage between rods blocked. The latter condition (complete fuel element flow blockage) would lead to coolant voiding and fuel element gross overheating or melting. Initial reactivity increase from such an accident is small, and results in a power change of only about 4%.

Gas entrainment in the primary sodium loop was considered for the passage of gas bubbles through the core, although the primary system has several design features to minimize gas entrainment. A series of small bubbles passing through the core would reduce core heat transfer, but could not insert enough reactivity to overcome Doppler feedback, However, a very large bubble, forming in the IHX and passing through the core center in the maximum sodium void worth region as tightly packed small bubbles, could give a reactivity insertion rate of ~80 $/sec. The response of the core to this latter accident would be similar to that for the loss-of-flow DBA.

For the containment evaluation, it is concluded that the containment barriers can be designed to withstand the consequences of all identified accidents which result in the maximum shock forces and pressure loadings. The inner containment structure can withstand the DBA resulting from an unprotected loss of flow. The outer containment structure can withstand a large sodium fire in an open vault. Potential radiation doses to the site exclusion radius from the unprotected loss of flow, the large sodium fire, and fuel handling accidents are shown to be well within suggested guidelines.

The economic optimization of the Task I preliminary reference design resulted in a modified design being selected for the Task III final reference design. A quantitative safety evaluation of the changes has not been made. Based on a qualitative evaluation, significant changes are not expected in the containment design for the DBA.

IV. ECONOMICS

Fast breeder reactors, as compared to light water reactors (LWR), have the potential for much lower fuel cycle costs. The major factors contributing to this potential are the absence of a requirement for a continuous supply of fissile feed material, the value of the net excess plutonium production, and the lack of reactivity burnup limitations. It is further anticipated that the fuel cost advantage for fast reactors will more than compensate for the probably higher capital costs. Several questions, listed here, are raised regarding these general statements:

1) How much lower are LMFBR fuel cycle costs likely to be, as compared to those of LWR, particularly during the early years of LMFBR introduction?

2) What is the probable differential in capital costs between a first generation commercial LMFBR and the then current LWR's?

3) Taking Items 1 and 2 into account, what must be the performance characteristics of an early LMFBR, particularly the fuel burnup level, in order for it to be economically competitive with LWR's during the 1980's?

The following is a brief outline of the approach taken to answer these questions:

1) Project an expanding combined LWR/LMFBR economy, based on a plausible rate of LMFBR introduction (Table IV-1)

2) Estimate the growth in fuel fabrication and reprocessing requirements, and plutonium production during the selected period

3) Couple these growing requirements with correlations of processing costs vs production level, to obtain trends in specific fabrication and reprocessing costs as a function of time (Table IV-2)

4) Estimate fuel cycle cost trends for both the reference LMFBR plant and a typical LWR, over their 30-year lifetime (Figure IV-1)

TABLE IV-1

NUCLEAR ECONOMY FORECAST

	1980	1990	2000
Installed Capacity (10^3 Mwe)			
Total Steam	550	940	1500
Nuclear (LWR plus LMFBR)	150	440	930
LMFBR	4	110	550
LMFBR Performance*			
Specific Fissile Inventory (kg/Mwe)†	3.8	3.0	2.5
Breeding Ratio	1.3	1.3	1.3
Compound Doubling Time (years)	13	10	8

*Projected improvement in LMFBR performance, based on oxide fuel

†Fissile Pu at mid-cycle equilibrium; in-core + ex-core

TABLE IV-2

PROJECTED TRENDS FOR LMFBR AND LWR FUEL CYCLE COST DATA

	1982-87	1987-92	1992-97	1997-2002
U_3O_8 Cost ($/lb)	8	9	10	12
Fissile Pu Value ($/gm)	10	11.2	11.7	12.5
LWR				
Fuel Fabrication Cost ($/kgU)	50	45	42	42
Fuel Recovery Cost ($/kgU)	30	27	25	25
LMFBR				
Fuel Fabrication Cost ($/kgH*)	225	170	150	130
Radial Blanket Fabrication Cost ($/kgU)	45	40	40	40
Axial Blanket Fabrication Cost ($/kgH)	35	30	30	30
Reprocessing Cost ($/kgH)	70	45	40	30
Fuel Ex-Core Holdup Time (days)	328	234	178	176

*kgH - kg of heavy element (U + Pu)

5) Calculate the LMFBR fuel cycle cost advantage, for various burnup levels, as a function of time; also assess the capital cost differential vs an LWR (Table IV-3)

TABLE IV-3

1000-Mwe PLANT CAPITAL COST SUMMARY*

Account	Description	($/kwe) LMFBR Cost	LWR Cost	Differential Cost (LMFBR-LWR)
21	Structures and Improvements	15.1	15.1	-
22	Reactor Plant Equipment	73.3	59.3	14.0
23	Turbine Generating Plant Equipment	28.2	39.4	(11.2)
24	Accessory Electrical Equipment	5.3	6.1	(0.8)
25	Miscellaneous Power Plant Equipment	1.1	1.1	-
TOTAL DIRECT CONSTRUCTION COSTS		123.0	121.0	2.0
	Indirect Construction Costs	16.3	16.3	-
	Owner's Contingency	6.0	3.6	2.4
	Interest During Construction	19.8	19.2	0.6
TOTAL CAPITAL COST (Rounded)		165.	160.	5.

Cost in ( ) implies penalty to LWR, relative to LMFBR

*Costs are based on January 1969 dollars, and do not include escalation to time of construction

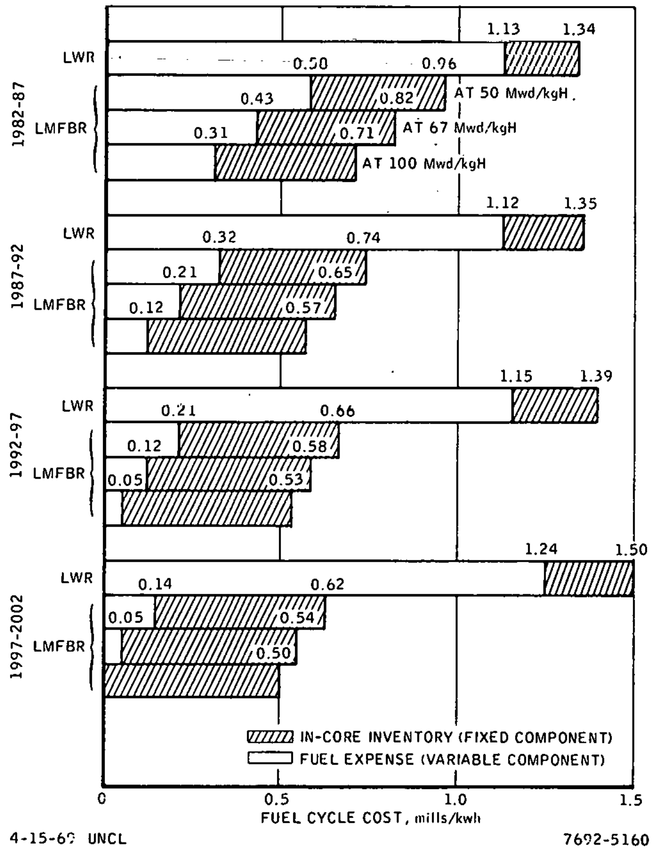

Figure IV-1. Comparison of LWR vs LMFBR Fuel Cycle Cost Trends, With LMFBR Burnup as a Parameter

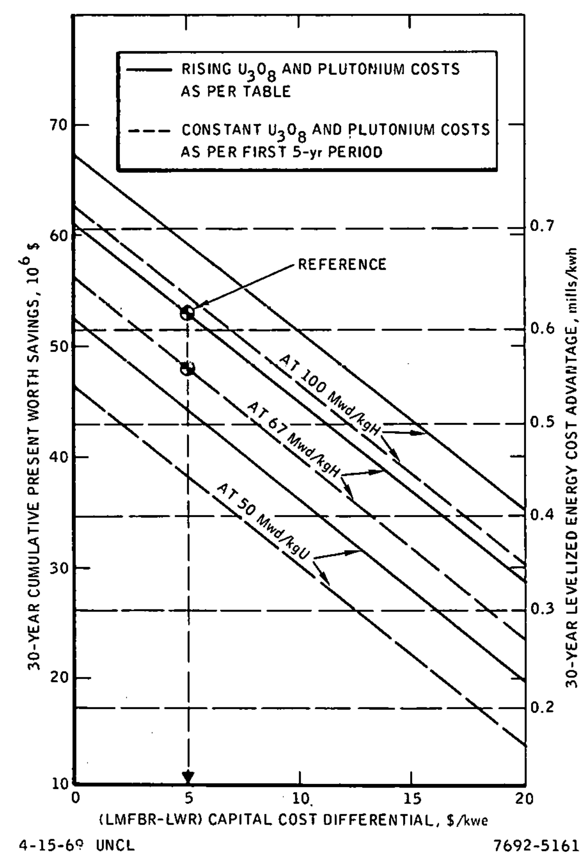

Figure IV-2. LMFBR vs LWR Competitiveness as a Function of LMFBR Burnup, Differential Capital Cost, and U_3O_8 Cost Trends

TABLE IV-4

30-YEAR DIFFERENTIAL (LWR-LMFBR) PRESENT WORTH ENERGY COST SUMMARY

5 Year Period (N)	Differential Fuel Cycle Cost (10^6 \$/yr)	Differential O & M + Nuc. Ins.* (10^6 \$/yr)	Differential Capital Fixed Charges (10^6 \$/yr)	Total Differential Power Generating Savings (10^6 \$/yr)	Cumulative Present Worth Differential Power Generating Savings (10^6 \$/hr)	Levelized Energy Cost Saving Thru Period N (mills/kwh)
1	3.64	(0.44)	(0.65)	2.55	10.45	0.364
2	4.90	(0.22)	(0.65)	4.03	22.23	0.451
3	5.67	-	(0.65)	5.02	32.69	0.513
4	6.72	-	(0.65)	6.07	41.71	0.563
5	6.72	-	(0.65)	6.07	48.14	0.592
6	6.72	-	(0.65)	6.07	52.72	0.628
30-year Total (Rounded)	\$172x10^6	(\$3x10^6)	(\$20x10^6	\$149x10^6	\$53x10^6	0.63

*Operation and Maintenance plus Nuclear Insurance
Cost in ( ) implies penalty to LMFBR, relative to LWR.

6) Assess the economic attractiveness of the reference LMFBR, by comparing the present-worth fuel cycle cost savings for various burnup levels with the estimated probable capital cost differential (Table IV-4 and Figure IV-2).

The major conclusion of the economic analysis is that an early (1981-83 on-line date) 1000-Mwe LMFBR, of rather conservative design and modest performance (Task III final reference design), is indeed economically attractive, offering an $\sim$ 10% energy cost saving, as compared to contemporary LWR's. This advantage is not particularly mitigated by various pessimistic assumptions, with respect to future ore costs or temperature-dependent LMFBR component and fuel technology.

A. NUCLEAR ECONOMY COST TRENDS

The anticipated trends for key fuel cycle cost factors are given in Table IV-2. These are based on the nuclear economy growth forecast shown in Table IV-1.

B. FUEL CYCLE COSTS

The fuel management program for the Task III final reference design is based on an annual refueling schedule, with one-half of the core and axial blanket, one-fifth of the inner radial blanket, and one-seventh of the outer radial blanket replaced at each refueling. For a plant capacity factor of 0.8, this schedule yields a core region average burnup of 67 Mwd/kgH for discharged fuel. A scatter reloading pattern is assumed, in which the new and old fuel elements are uniformly distributed within each region.

The fuel in the core region consists of a mixture of depleted uranium (0.2% U^{235}) and plutonium oxides. The isotopic composition of the feed fuel plutonium was 67% Pu^{239}, 26% Pu^{240}, 5% Pu^{241}, and 2% Pu^{242}, as established by contract ground rules. The blanket feed was depleted uranium (0.2% U^{235}).

The in-core fissile plutonium inventory at equilibrium midcycle is 2672 kg; and the ex-core inventory, based on the ex-core times for the initial 5-year period, is 1121 kg. Thus, the specific inventory is 3.79 kg/Mwe, and the compound doubling time is 13 years.

The estimated equilibrium fuel cycle costs for the Task III final reference design, at the reference burnup level of 67 Mwd/kgH, are shown in Figure IV-1 for each of the four 5-year periods. Figure IV-1 highlights the facts that:

1) LMFBR fuel cycle costs decrease sharply during the initial 10 years of operation.

2) LMFBR plants will enjoy very low incremental operating costs (fuel expense) — 0.7 mills/kwh lower than LWR, in early life. The low incremental cost for the LMFBR should favor the plant in the loading schedule. The assumption of 0.8 capacity factor for both the LWR and LMFBR conservatively understates the economic benefits of the LMFBR.

C. CAPITAL COSTS

The estimated capital costs for the Task III final reference plant (Table IV-3) were obtained from an estimate of the engineering and construction costs. A comparable breakdown of the capital cost of a typical LWR is also shown on this table. These costs are based on information available as of January 1969, and do not include escalation to the assumed date of actual construction. The utility owner has been considered as the prime contractor, using the services of an architect-engineer or an engineer-constructor. In either case, the prime contractor will procure the nuclear steam system and associated engineering services from a reactor supplier. The reactor supplier's overheads, contingencies, and profit are included in the direct construction costs.

The plant is assumed to be the first unit on a new river site, with characteristics similar to the AEC hypothetical site described in TID-7025. A basic "ground rule" for this estimate is that the plant will be one of a series of plants, and that first-of-kind engineering costs have been amortized over previous plants.

The indirect construction costs shown were patterned after the costs given in WASH-1082 for an LWR plant. This was done to facilitate comparison between the two plant types.

Referring to Table IV-3, we note that the estimated total capital cost for the LMFBR is \$165/kwe, whereas the total capital cost for a comparable LWR plant is \$160/kwe. Note that the LMFBR enjoys an ~\$11/kwe cost advantage in the power conversion portion of the plant, because of its better steam conditions; whereas it suffers a \$14/kwe penalty in the reactor portion of the plant, mainly because of the added costs of the intermediate coolant loop.

D. COMPETITIVENESS OF THE LMFBR

Figure IV-2 shows the competitive advantage and sensitivity of the LMFBR to economic and fuel performance variations. This is based on a 30-year present worth analysis (at a 7% factor) of the LMFBR and the LWR. The principal conclusions which may be drawn from these results are as follows:

1) There is very little difference in the competitiveness of the LMFBR relative to an LWR, whether one assumes a constant U_3O_8 cost of \$8/lb or a rising U_3O_8 cost trend.

2) The reference LMFBR plant offers ~10% (0.45 mills/kwh) saving in power generation costs over an LWR, even at a constant U_3O_8 cost of \$8/lb and twice (\$10/kwe) the estimated capital cost differential.

3) The indicated competitive margin for the reference LMFBR plant design, coupled with results of the change in energy costs with reactor outlet temperature, suggest that early plants need not go to the reference 1140°F outlet temperature, and its attendant more critical material demands, in order to be competitive. For example, an 80°F drop in reactor outlet temperature, from 1140° to 1060°F, results in about a \$6/kwe capital cost increase, or a 0.1 mills/kwh energy cost penalty. As can be seen from Figure IV-2, this cost penalty would still leave the LMFBR with at least a 0.3 mills/kwh energy cost advantage over an LWR.

E. AVAILABILITY

The large investment in power stations consisting of single units with generating capacities > 1000 Mwe, coupled with the ability of a breeder to produce excess fuel, causes the unit's "on-line" capability to have a very large economic effect on its commercial worth. A definition commonly used to describe the availability of conventional steam plants is "operating availability," defined* as

*"Report on Equipment Availability for the Seven Year Period 1960-66," EEI Publication No. 67-23 (August 1967)

the percentage of time a unit is available for service, whether operated or not. This "time basis" availability is not completely suitable for the 1000-Mwe LMFBR, since the multiloop design presents the possibility of a single-loop forced outage (forced power reduction to 70% of nameplate). With a time basis definition, no availability penalty would result from this single-loop outage. Plant energy availability has therefore been used in this study. It is defined as the percentage of rated plant energy which the mature plant is capable of producing. For example, if the 1000-Mwe plant is capable of producing 9,000,000 Mwh in a 10,000 hr period, the availability would be 90%.

The plant energy availability goals for the Task III final reference design are tabulated in the following listing:

Item	%
Overall availability	88.9
Forced outage rate	6.4
Scheduled downtime for turbine and concurrent maintenance	4.7

APPENDIX
GROUND RULES AND DESIGN BASES

The basic Follow-On Study guidance was established by ANL. Additional ground rules, required for the study and not specified by ANL, were established by AI. The major guidance specified by ANL, and the AI ground rules, are summarized in the following sections.

A. GENERAL

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 art in 1980.*

The plant shall produce 1000 Mwe, net.

The reactor heat transfer medium shall be liquid sodium.

The fuel requirements shall be based on an equilibrium plutonium-uranium fuel cycle.

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).

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.

Fuel doubling times shall be computed on a compound interest basis, including total system fuel inventory, and be studied within the range of 7 to 15 years.

B. SITE

The site assumed for the plant will be one suitable for a central station nuclear power plant of this size and type, having good soil conditions, adequate

*See Section E of this appendix for interpretation of "State of Art in 1980" water availability, adequate undeveloped acreage, reasonable stream and air pollution ordinances, accessibility by highway, railroad, and navigable water systems, and other features commonly sought for the installation of such plants.

The site will be assumed to be such that no unusual design features are required because of local conditions, such as seismic zoning, peripheral land use, nearby habitation, and unfavorable meteorology.

The site will be assumed to be a new site, and to require all of the component services and facilities inherent to a first-unit installation.

Containment design, and related design features, shall be based on population distribution data stated in TID-7025, Vol. I., Section 110.

C. OPERATION

The plant shall be designed to meet the utility companies' base load system demands.

D. SAFETY REQUIREMENTS

The nuclear power plant shall be designed in accordance with the AEC's "General Design Criteria for Nuclear Power Plant Construction Permits," dated July 11, 1967, and all subsequent modifications and interpretations, except where the contractor can show that the criteria are either inapplicable or not stringent enough. When the AEC criteria are not to be followed, the contractor shall indicate his reasons for not following them; and he shall substitute his own criteria where he feels that the intent of the AEC criteria can be met with his criteria. This requirement shall not be construed to preclude the contractor from establishing other design criteria which he feels should be required of large sodium-cooled fast breeder reactors.

E. STATE OF ART IN 1980

The terms, "1980 technology" and "state of art in 1980", shall be interpreted as referring to a target date in the period 1980 to 1990 for initial operation of a 1000-Mwe LMFBR power plant. Assuming a five-year construction period, this would place the date of sale of the plant to the utility in the period 1975 to 1985. The selected sale date is to be specified by the contractor.

The Initial Reference Design will be established, using the subcontractor's best judgment as it applies to the probable availability of necessary components.

F. DESIGN

The purpose of the Ground Rules is for meaningful comparisons among the studies. The contractor shall adhere to the following Ground Rules:

1) The breeding ratio shall be defined as the ratio of the rate of destruction of fissile atoms, by capture and fission. This ratio shall be taken over the entire reactor, and at the midpoint of a burnup cycle, for the equilibrium core and blanket fuel cycle. (For the purpose of determining breeding ratio only, the Pu^{239}, Pu^{241}, and U^{235} shall be considered fissionable.)

2) The reactor fuel cycle cost shall be evaluated, based upon equilibrium conditions. The plutonium isotopic composition in equilibrium cycle replacement fuel shall be assumed to be:

   Pu^{239} - 67 at.%

   Pu^{240} - 26 at.%

   Pu^{241} - 5 at.%

   Pu^{242} - 2 at.%

3) Sodium void and Doppler coefficients shall be calculated at midcycle of the equilibrium core.

4) The primary and secondary coolant shall be sodium. Sodium properties, given in ANL-7323, including subsequent additional and modifications, shall be used. ANL-6246 shall be used for data not covered in ANL-7323.

5) The following codes, standards, and regulations shall apply, where applicable:

   United States Atomic Energy Commission — Code of Federal Regulations — Title 10

   ASME Section I — Power Boilers

ASME Section II — Materials Specifications

ASME Section III — Nuclear Vessels

ASME Section VIII — Unfired Pressure Vessels

ASME Section IX — Welding Qualifications

ASTM — American Society for Testing Materials

ASA — American Standard Association

AISC — American Institute of Steel Construction

TEMA — Tubular Exchanger Manufacturers Association

NEMA — National Electrical Manufacturers Association

NFPA — National Fire Protection Association Publication

ACI — American Concrete Institute

AWS — American Welding Society

ISA — Instrument Society of America

AEC Cost Evaluation Handbook (for direct construction costs), where applicable.

In the event of conflict between any of these codes and current technology, the most stringent interpretation shall apply.

G. ECONOMICS

Installed capital cost estimates shall be reported in accordance with the AEC's Classification of Construction Accounts — Nuclear Power Plants.

For capitalizations use:

Plant capacity factor, 90% of availability factor

Annual capital charge, 13%

Plant availability factor equals 100%, less downtime for refueling.

Fuel cycle cost data shall be applied as follows:

Fissile Pu value	$10/gm

Charges on fissile Pu in core, blanket, reprocessing, refabrication, and shipping	10%/year
Reprocessing plant Pu criticality limit	67% of U^{235} limit
Reprocessing plant charge	$30,000/reprocessing day*
Reprocessing plant throughput, as function of U^{235} concentration	Determined from TID-7025
Shipping cost, core and axial blanket	$20/kg
Shipping cost, radial blanket	$10/kg
Fabrication cost	Adjusted for design
Fuel fabrication charge	10% of book value
Nitrate conversion to feed material	Adjusted for design
Fabrication losses	1% (included in Pu credit)
Reprocessing losses	1% (included in Pu credit)

For an option between design features in a plant, a suggested worth of availability on an average basis is $2000/hr for a 1000-Mwe plant.

H. ADDITIONAL AI GROUND RULES

Additional ground rules for the Follow-On Study which were established by AI are as follows:

1. Site

The assumed site lies along the bank of a navigable river. The land is relatively flat, clear of existing structures, and is above flood level. The ground is easily excavated, down to a depth of 100 ft; no blasting is required.

Good secondary roads and a railroad spur are available at the site boundary.

Site temperatures, for design purposes, range from 90°F dry bulb in the summer to 0°F dry bulb in the winter. Cooling water is available at 55°F, in

*An allowance to cover the expense of turn-around time is included in the $30,000 figure. This cost is associated with 150 days of cooling before reprocessing.

a supply adequate for all plant cooling requirements, including main condenser cooling.

2. Economics

For purposes of computing present worth of future savings, a present worth factor of 7% was used.

A plant factor of 0.8 was used; this was based on a plant capacity factor of 0.90 and an availability factor of 0.889.

Cost estimates were based on January 1969 values.

3. Plant Design

The IHX's and steam generators are to be elevated; so that, in the event of a reactor trip, sufficient core cooling is provided by free convection sodium flow in the primary and secondary loops.

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