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CONCEPTUAL DESIGN FOR A 75 MWe MIXED
SPECTRUM SUPERHEATING REACTOR
POWER PLANT

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February 25, 1962

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1.0 INTRODUCTION

1.1 Scope

This report presents the conceptual design of a 75 MWe prototype Mixed Spectrum Superheater power plant. The scope of the work has emphasized primarily the design, performance, and cost information on the nuclear portion of the plant. The research and development programs required to insure plant feasibility are also presented.

This work was performed under Task J of the Nuclear Superheat Project in partial fulfillment of the requirements of Contract AT(04-3)-189, P. A. #13.

1.2 Background

Conceptual design work on the Mixed Spectrum Superheating Reactor (MSSR) was originally performed by the General Electric Company under the company-sponsored Advance Engineering program in the Atomic Power Equipment Department. Results of this work indicated that the Mixed Spectrum Superheater might have significant advantages over alternate nuclear superheating concepts being investigated under Atomic Energy Commission programs by General Electric and other companies. It was therefore proposed to the AEC that the work on superheat already being performed by General Electric for the Commission be extended to include a more detailed conceptual design and feasibility study of the MSSR. The AEC authorized the beginning of such a study in April of 1960.

The work performed under this study supported the earlier conclusion that the Mixed Spectrum Superheater concept is feasible and economically attractive. Effort was therefore applied to the conceptual design and evaluation of a 300 MWe reference design. The results of this work, reported in GEAP-3590, "Steam-Cooled Power Reactor Evaluation, Mixed Spectrum Superheater", proved to be sufficiently favorable so that work on the concept was continued.

As part of the long-range development program proposed for the MSSR, it was recommended that a small reactor prototype precede the design and construction of a large demonstration plant of the type described in GEAP-3590. Ideally, the prototype would provide, with minimum financial risk, a complete test of the reactor concept. This would include development of information on the operating physics of the reactor, on fuel element performance, on the stability and reliability of the reactor, and on costs of components. After the completion and initial operation of the reactor experiment, the design of the large demonstration plant could proceed with confidence.

1.3 Design of the Prototype

The design of a suitable prototype for the Mixed Spectrum Superheater

presents some unusual problems. The physics characteristics of the system are strongly size dependent. As the size of the superheating region is made small, the fissionable material concentration (enrichment) goes up, and the neutron spectrum and physics characteristics of the core change. Thus, as the size departs radically from that of the large power plants, extrapolation of the physics data becomes less sure. Perhaps more important is the effect on the Doppler coefficient in the superheating section. As the core is made smaller, the magnitude of the negative Doppler effect decreases. Since the Doppler effect is the only reliable fast shutdown mechanism in the superheating section, a low value may compromise the safety of the prototype reactor. These considerations, in effect, set a minimum range for the physical size of the superheating core. This size has been taken as 30", which results in a Doppler effect about one-half of that of the large reference design reactor and nuclear characteristics sufficiently similar to a large power reactor so that it is believed extrapolation can be made with confidence.

Other design parameters of the prototype reactor have been arrived at on the basis of setting operating temperature levels identical to those in the reference design plant. In particular, the outlet steam temperature has been set at 950°F and the maximum clad surface temperature at 1250°F. In addition, in order to obtain a fuel temperature distribution within the fuel element characteristic of that in the large plant, the fuel has been sized so as to operate at the same central temperature. This latter point is important for evaluation of the Doppler effect in the operating reactor.

Analysis shows that the requirements outlined above dictate a minimum power level. This level is determined by the minimum spacing between fuel pins which can be tolerated in the core. For the present work, a minimum pitch-to-diameter ratio of 1.05 was chosen, which results in operation in a region of heat transfer and fluid flow which has not been well explored. The power level resulting from the considerations above is 75 MWe.

There is some freedom to reduce the power level below this value by making use of a two-pass flow scheme in the superheating section, but this leads to a mechanical design and to operating procedures which depart significantly from that of the proposed large scale plant. A description of a lower powered two-pass prototype is given in GEAP-3877*.

GEAP-3877 - Nuclear Superheat Project, Ninth Quarterly Progress Report, October-December, 1961.

It should be noted that in one respect, the 75 MWe prototype does not reproduce the operating performance of fuel in the large power reactor. This involves the power density in the fuel and maximum heat flux at the clad surface. To simulate these characteristics would require operation at over twice the 75 MWe power level. It was felt that the high specific power capability of the fuel could be demonstrated elsewhere and that the prototype should be designed to prove the basic feasibility of the Mixed Spectrum concept at the minimum cost. High-specific-power fuel of the type proposed for use in the superheat section is presently being developed under other AEC programs.** Assuming success of these programs, the major unknown will be the performance of the fuel clad in the fast neutron spectrum superheat environment of the MSS. As noted above, fuel clad operating conditions have been chosen to simulate those of the large power reactor.

In arriving at the plant design presented in this report, primary emphasis was placed on insuring experimental flexibility and the ability to simulate the characteristics of the large Mixed Spectrum power reactor. No attempt was made to produce a minimum cost 75 MWe plant or to minimize its fuel costs.

In terms of experimental flexibility, several features are of particular interest. First, the design is such that it should be possible to increase the power level of the reactor to 100 MWe with essentially no changes in the plant except provision for extra turbine capacity and an additional heat dump. The plant can also be modified to operate as a forced-circulation, 150 MWe plant. To provide for this possibility, nozzles are included on the vessel and there are provisions for piping, pumps and other equipment in the plant design. At the 150 MWe level, the power density in the core and the fuel performance approach that of the 300 MWe reference design.

Finally, the superheating section in the center of the core has been designed so that the entire assembly can be removed and the reactor operated as a saturated boiling water reactor. This latter feature is aimed at reducing the financial risk involved in testing a new concept, since the boiling reactor resulting from the removal of the superheat section would have about the same operating costs as a standard BWR.

1.4 Plant Design and Required Development

The design of a reactor in sufficient detail to obtain good costing information represents a very large undertaking. Fortunately, the experience of General Electric in the design and construction of boiling reactors could be relied on for the present work. In areas

** GEAP-3833-AEC. The Post-irradiation Examination of a $\text{PuO}_2\text{-UO}_2$ Fast Reactor Fuel.

external to the reactor, the design utilizes standard BWR practice with modifications made to factor in the special requirements caused by the addition of superheat. It was thus possible to arrive at a rather detailed plant description while placing the major design effort on the special characteristics of the Mixed Spectrum Superheater.

It should be pointed out that considerable development effort is required before operation of this plant described herein can take place. Where possible in the text, specific areas of major development are called out. However, in order to make the report more readable and avoid descriptions in terms of several alternatives, the plant is sometimes discussed in concrete terms, whereas the actual design is dependent upon the results of the development program.* The reader is directed to Section 15.0, which specifically calls out the major areas of required development recognized today.

* A specific example is the choice of Hastelloy-X as the reference fuel material. At present, the General Electric Company, under the AEC Superheat program, is engaged in an extensive effort aimed at superheat clad development. Hastelloy-X is only one of a number of materials being tested, and it is quite possible that it will prove unsuitable or that a different material will prove superior. Indeed, at the present time there is no material which has been demonstrated to be suitable under the conditions postulated for the superheater.

2.0 SUMMARY

2.1 Description of Plant and NSSS

The Mixed Spectrum Superheating Reactor (MSSR) is a nuclear reactor contained within a single pressure vessel in which both boiling of water and superheating of steam take place. The reactor core consists of four annular regions, which, in going from inside to outside, are: an unmoderated region (fast core), an unmoderated buffer region (fast buffer), a water-moderated buffer region (thermal buffer), and a water-moderated boiling region (thermal core). The boiling region is very similar to that in a conventional boiling water reactor core. Both moderated regions are cooled by water; the unmoderated regions are cooled by steam.

The neutron spectrum varies from a thermal spectrum in the boiling part of the core to a fast spectrum in the superheating region. Thus, the designation of Mixed Spectrum Superheating Reactor is applied to the present concept.

The basic features of the MSSR are shown in Figure 3.1.1 (Drawing No. 104R674). Water is boiled in the outer boiling core region. The steam-water mixture passes up through the chimneys above the fuel elements to the free surface where the steam is separated. The steam passes through the dryers and then travels down through an offset duct to the unmoderated superheating core. Water surrounding the offset duct reduces radiation streaming to the vessel head to minimize activation of the head. After passing through the superheating core, the steam leaves the pressure vessel through the pipe that penetrates the vessel bottom head. Beneath the superheating core within the fast core shroud is a series of shielding plugs which reduce radiation streaming to the vessel bottom.

The boiler is cooled by natural circulation. The water separated from the steam flows downward outside the chimneys where it mixes with 300°F feedwater. The subcooled water then flows downward between the boiler shroud and the pressure vessel to the bottom of the vessel where it turns and flows upward through the boiling core. Nozzles have been provided in the present vessel for later conversion to forced circulation if higher power operation is desired.

The reactor is pressurized to 1015 psia. Superheated steam is supplied to the turbine at 900 psia and 950°F. Feedwater is returned to the vessel at about 300°F. A reactor thermal power of 206 MW of which 23.5 per cent is supplied by the superheating core is required to produce 75 MW of gross electrical power. The gross cycle efficiency is estimated to be 36.4 per cent.

The boiler power is controlled manually by bottom-mounted poison elements. The superheating core power and the thermal buffer are controlled by top-mounted control drives. In addition, coupling control poison blades which are bottom-mounted are provided between the fast and thermal cores. System steam pressure is regulated automatically by the turbine control and by-pass valves. The superheater steam outlet temperature is

indicated with thermocouples and controlled manually by manipulating the control elements in the fast core. Load rejection is absorbed by the use of full-capacity by-pass valves to the condenser.

2.2 Reactor Building

The reactor building, shown in Figures 2.1.1 and 2.1.2 (Drawing No. 142F239), contains the Nuclear Steam Supply System. This includes fuel transfer equipment, fuel storage room for new fuel, pool storage space for spent fuel, fuel pool cooling and cleanup system, emergency condenser, emergency steam dump tank, shutdown heat exchanger system, emergency liquid poison injection system, and reactor service cooling water system. A 60-ton capacity polar crane is provided for reactor servicing, fuel handling, and general maintenance. The reactor and its radioactive components are contained within a vapor-tight, steel enclosure. The building is designed to withstand the effects of a maximum credible accident which would yield an estimated peak pressure of 30 psig. It is felt that future versions of the MSSR will be able to take advantage of pressure suppression as a means of containment. In the present plant, however, where the primary purpose is to demonstrate feasibility of the concept as well as to establish plant characteristics, it is felt that dry containment is more appropriate. Two air locks, one for personnel and the other for equipment, are provided in the building.

Inside the reactor building, much of the structures are reinforced concrete which also serves as shielding where required. Space is provided within the building for the addition of recirculation pumps and other equipment necessary to double the capacity of the system.

2.3 Balance of Plant

Primary emphasis in this study has been placed on the Nuclear Steam Supply System. The design of the Balance of Plant does not differ greatly from conventional plant practice. Only sufficient information is generated to provide a basis for the cost estimate.

2.4 Plot Plan

The conception of the general layout, as shown in Figure 2.1.3 (Drawing No. 142F239, SH3), is based upon the following main principles:

1. Interlocation of buildings according to function.
2. Accessibility of reactor building, turbine hall, and loading dock by road and railway.
3. Possibility of exercising access control to controlled areas, i.e., to reactor building and part of the turbine hall.
4. Future expansion to double the generating capacity of the plant.

To provide the shortest possible length of steam and feedwater connections, the reactor building is placed adjacent to the turbine hall but at the opposite end of the building from the turbine. In this position, it is out of line from the rotating part of the turbine, thus minimizing the possibility of damaging the reactor building by a turbine accident. It will also be between the two turbines, if a second unit should be added later.

Heating and ventilation facilities are placed in the end of the turbine building nearest the reactor building. The stack is next to the reactor building near the ventilation room. The rad waste facility is located next to the river in the vicinity of the reactor building and turbine hall.

Office building and control room are located adjacent to the turbine hall near the reactor building. The present plot plan also shows an access control building connected with the office building. This building and the control room may be combined with the office building, since the space shown is probably more than adequate. A warehouse and shop are also provided.

Primary Coolant System

The piping and instrumentation layout for the Nuclear Steam Supply System is shown in Figure 2.1.4. The corresponding piping specifications are given in Figure 2.1.5.

2.5 PLANT PERFORMANCE DATA SUMMARY

Summary of Reactor Plant Characteristics

<u>Description</u>	<u>Superheat</u>	<u>Boiler</u>	<u>Total Plant</u>
A. Heat Balance			
1. Net Plant Power, MW(e)			71.6
2. Total Reactor Power, MWt	48.5	157.5	
B. Turbine Cycle Conditions			
1. Throttle Temperature, °F			950
2. Throttle Pressure, psia			930
3. Total Steam Flow, lb/hr			575,000
C. Reactor Description			
1. Reactor Vessel			
a. Inside Diameter, ft.			11
b. Inside Height, ft.			37.3
c. Wall Thickness, in.			4.75
d. Design Pressure, psig			1250
e. Design Temperature, °F			575
2. Reactor Core			
a. Equivalent Diameter, in.	36	93	
b. Active Height			
Superheater/Boiler, in.	30	80	
Buffer, in.	54*	38	
Reflector, in.	12		
c. Active Core Volume (Incl. Buffer), ft ³	17.8	297	
d. Fuel Loading			
Superheater, Kg U	1670		
Boiler, Kg U		14,700	

* Including reflector

<u>Description</u>	<u>Superheat</u>	<u>Boiler</u>	<u>Plant</u>
2 d. (Contd.)			
Buffer, Kg U	820	1,750	
Reflector, Kg U	1,390		
e. Enrichment			
Superheater, a/o U-235	26.3		
Boiler, a/o U-235		3.0	
Buffer, a/o U-235	Natural	1.25 - 1.55*	
Reflector, a/o U-235	Depleted (0.3)		
f. Structural Material	SS	Zr-4	
g. Neutron Moderator	None	H ₂ O	
h. Moderator to Fuel Ratio			
Boiler		2.6	
3. Reflector			
a. Material	Depleted U	H ₂ O	
b. Axial Thickness, ft.	1.0	4 +	
c. Radial Thickness, ft.		1.3	
4. Fuel Elements (for each type)			
a. Fuel Material	UO ₂	UO ₂	
b. Fuel Element Geometry Pitch Arrangement	Rods Triang.	Rods Square	
c. Clad Material	Hastelloy "X"	SS	
d. Fuel Material Outside Diameter			
Superheater/Boiler, in.	.378	.511	
Buffer, in.	.378	.511	
Reflector, in.	.378		

* See Section 3.5.2.2 for detailed description

<u>Description</u>	<u>Superheat</u>	<u>Boiler</u>	<u>Total Plant</u>
4. (Contd.)			
e. Clad Thickness			
Superheater, in.	.030		
Boiler, in.		.013	
Buffer, in.	.030	.013	
f. Fuel-Clad Gap (cold), in.			
(Equilibrium)	0	0	
g. End Plenum Length			
Superheater/Boiler, in.	3	3	
Buffer, in.	3	3	
h. Gap Filler Material	He	He	
5. Fuel Assemblies			
a. Number of Assemblies			
Superheater			
Fast Core Control	4		
Fast Core	60		
Fast Buffer, corner	4		
Fast Buffer, side	28		
Boiler			
Thermal Core		128	
Thermal Buffer		60	
b. Number of Rods/Channel			
Superheater			
Fast Core	52		
Fast Core Control	42		

<u>Description</u>	<u>Superheat</u>	<u>Boiler</u>	<u>Total Plant</u>
5 b. (Contd.)			
Fast Buffer, corner	42		
Fast Buffer, side	26		
Boiler			
Thermal Core		36	
Thermal Buffer		36	
c. Cross Section Dimensions (inside)			
Superheater			
Fast Core, in.	3.269 x 3.243		
Fast Core Control, in.	3.010 x 2.872		
Fast Buffer Corner, in.	5.080 x 1.675		
Fast Buffer Side, in.	3.269 x 1.675		
Boiler			
Thermal Core, in.		4.849 x 4.849	
Thermal Buffer, in.		4.849 x 4.849	
d. Lattice Spacing, in.	3.469	11	
e. End Fitting Material	SS	SS	
6. Reactor Control			
a. Method of Control	Fuel	Poison	
b. Absorber Material		B ₄ C	
c. Number of Control Elements			
Superheater/Boiler	4	36	
Buffer		16	
Coupling Control		8	

<u>Description</u>	<u>Superheat</u>	<u>Boiler</u>	<u>Total Plant</u>
6. (Contd.)			
d. Cross-section Dimensions			
Superheater (square shaped), in.	3.210 x 3.210		
Boiler (cruciform), in.		9.5 x 9.5	
Buffer (cruciform), in.		8.688 x 8.688	
Coupling Control (flat plate), in.		11	
e. Effective Length			
Fuel/Poison Section, in.	30	80	
f. Type of Drive			
Superheater	Twin Screw		
Thermal Core		Locking Piston	
Thermal Buffer		Twin Screw	
Coupling Control		Locking Piston	
7. In-Core Flux Monitors			
Number	4	24	
Number per Channel	1	3	
Number of Channels	4	8	
D. Performance Data			
1. Reactor Coolant	Steam	Water	
2. Reactor Coolant Outlet Temperature, °F	950	545	
3. Reactor Coolant Inlet Temperature, °F	545	539	
4. Primary System Operating Pressure, psig	1000	1000	
5. Primary Coolant Flow, lbs/hr	5.75×10^5	10.7×10^6	

<u>Description</u>	<u>Superheat</u>	<u>Boiler</u>	<u>Total Plant</u>
D. (Contd.)			
6. Ave. Core Coolant Velocity, ft/sec	108	4	
7. Max. Fuel Center Temp., °F (Based on $k = 1.3$ Btu/hr ft °F)	4320	3400	
8. Max. Cladding Surface Temp., °F	1230	560	
9. Burnout Safety Factor		2.85	
10. Max. Core Heat Flux, Btu/hr ft ²	386,000	290,000	
11. Ave. Core Heat Flux, Btu/hr ft ²	156,700	86,300	
12. Ave. Core Power Density, kwt/ft ³	2,810	598	
13. Peak to Average Power Ratio	2.46	3.36	
14. Ave. Specific Power, kwt/kg U	22.8	10.3	
15. Fuel Management	4 Batch	5 Batch	
16. Ave. Fuel Burnup, * MWD/MT	50,000	15,000	

* These numbers do not include fast or thermal buffers.

E. Suggested Containment

1. Design Criteria	ASME Code
2. Type	Dry
3. Primary Loop Coolant Inventory, lbs	75,000
4. Geometry	Cylinder
5. Diameter, ft	72
6. Design Pressure, psig	30
7. Material	ASTM A-201 Grade B

2.6 Development Programs

This section summarizes the development program presently envisioned to carry out the design and construction of the 75 MWe prototype plant. A more detailed discussion of the proposed development program is given in Section 15.0 of this report.

2.6.1 Fuel Element Development

a. Cladding Materials:

Various materials must be investigated to develop a suitable cladding and other materials for operation at high temperatures in superheated steam. Primary interest centers on the high-nickel-alloys such as Incoloy, Inconel, and Hastelloy, although other materials are also being considered. Only limited data are available at present on the irradiation and stress corrosion behavior of these alloys.

b. Fuel Design and Fabrication:

The superheater fuel element must be analyzed in detail considering such design parameters as plenum voids and UO_2 -clad interaction. When choice of cladding material has been made, its creep behavior and strength characteristics must be factored into the fuel design. Practicable means for accurate spacing of fuel rods must be developed.

Manufacturing techniques for assembling fuel rods into channels for the superheat fuel bundles must be developed. Incorporating thermal and epithermal poisons into the fuel and fuel channels requires further study and experimentation. Irradiation and burnup data, particularly with regard to fuel and poison migration, will be required.

c. Irradiation Tests:

Prototype fuel elements should be irradiated in-pile under conditions simulating the proposed operating steam conditions. These tests should be supplemented by capsule irradiations of small fuel samples.

d. Heat Transfer Tests:

The effects of close packing (low pitch-to-diameter ratios) on heat transfer must be more firmly established. Of particular interest are the peripheral variation in surface temperature as a function of rod spacing. Knowledge of the degree of flow mixing is also desirable.

e. Flow and Vibration Tests:

An attempt will be made to explain quantitatively the parameters that influence the vibrations of multi-rod assemblies in parallel flow. The data will be analyzed to relate results to various reactor operating conditions to assist in the design of fuel rod spacers. Average pressure drop measurements will also be made for various rod configurations.

2.6.2 Emergency and Shutdown Spray Cooling

Spray cooling of the superheating core plays an important part not only when normal steam cooling capability is inadvertently lost, but also during normal startup and shutdown of the MSSR. Major unknown area to be explored is the effectiveness of spray cooling in preventing overheating of closely spaced fuel rods under the foreseeable operating conditions. Information must also be developed to provide a basis for establishing the optimum time for turning the spray cooling on and off during reactor shutdown and startup.

2.6.3 Mechanical Development

a. Fast Core Shroud and Supports:

The shroud separating the superheating core from the boiler depends upon the fast core for support to withstand the pressure drop between the two sections. The shroud therefore collapses against the fast buffer under operating conditions. The major foreseeable problems to be investigated are:

1. Achieving a symmetrical deformation so as to not upset alignment.
2. Keeping stresses sufficiently low to achieve high reliability and prevent buckling.
3. Making transition sections.
4. Determining the effects of thermal stresses and thermal and mechanical cycling.

Analytical studies, mock-ups, and model tests are anticipated.

b. Fast Core Control Element Guidance:

The flexible wall of the superheating core shroud may cause some misalignment of the control element guide tubes in the fast core. Misalignments due to thermal gradients could also contribute to this problem. The coupling between the control drive shaft and the control fuel assembly must be such that

free movement within the guide tube is assured under all anticipated conditions. Various alternate designs with model tests will be required.

c. Fast Core and Thermal Buffer Control Drive Design:

The control elements in the fast core and thermal buffer are activated by control drives mounted on the top head. It is planned to use the General Electric twin-screw drives currently in use on small BWR's. Adaptation of these drives to the MSSR will require some development effort, particularly with regard to the high-pressure seals at the vessel penetrations.

d. Steam Outlet Nozzle Sliding Seal:

Because of differential thermal expansions, the steam outlet pipe is not rigidly joined to the pressure vessel. As an alternative to a bellows to accommodate this expansion, the present design uses a sliding seal arrangement comprising three concentric pipes with piston-ring type of seals. Seals of this type in the sizes required (18 inch diameter pipe) are uncommon but not unknown. A modest development effort supplemented by tests on various configurations is anticipated to insure reliable operation and reasonable accessibility for maintenance.

e. Other Mechanical Design Development

In addition to the specific problem areas described above, it is anticipated that other design difficulties will be revealed as the detailed design is undertaken. In this category will be such items as special valves (e.g., valve for the emergency cooling system), refueling devices, etc.

2.6.4 Reactor Stability Studies

This phase of the development program will supplement the BWR stability studies now in progress at APED. The objective will be to identify, evaluate, and analyze the stability problems unique to the Mixed Spectrum Superheater to assure that the addition of the fast core does not lead to instabilities. Theoretical studies and analyses will guide the programming of reactor tests to be carried out following reactor startup.

FIGURE 2.1.1

DRAWN BY <i>J. Mastromarino</i>	APPROVED		GENERAL ELECTRIC ATOMIC POWER EQUIP. DEPT. SAN JOSE, CALIF.
	BY	FOR DATE	
CHECKED BY <i>J. Mastromarino</i>	<i>K. L. L.</i>	<i>11/15/63</i>	TITLE REACTOR BLDG ARRGT. ELEV.
ENGINEER <i>J. Mastromarino</i>			MADE FOR MSSR 75MW PROTOTYPE
			SCALE 1" = 1 FT - 0"
			DWG. NO. 142F239
			SH.1
			REV.

PLAN "CC"

PLAN "EE"

PLAN "GG"

PLAN "DD"

PLAN "FF"

FIGURE 2.1.2

SEE SH.1 FOR ELEVATIONS.

DRAWN BY <i>J. M. H. / 10-10-62</i>		APPROVED BY FOR DATE <i>K. H. / 11-1-62</i>		GENERAL ELECTRIC ATOMIC POWER EQUIP. DEPT. SAN JOSE, CALIF.	
CHECKED BY <i>J. M. H. / 1-10-62</i>				TITLE REACTOR BLDG ARRGT - PLAN	
ENGINEER <i>J. M. H. / 1-10-62</i>				MADE FOR MSSR 75 MWe - PROTOTYPE	
155000		11		SCALE 1" = 10'-0"	DWG. NO. 142 F 239 SH. 2 REV.

149A4222		SH NO. 1				
REV. NO. 149A4222	TITLE Proposal PIPE MATERIAL	CONT ON SHEET	SH NO. 1			
CONT ON SHEET	SH NO. 1	FIRST MADE FOR MCAP 3004-B (75 MW)				
FIGURE 2.1.5			REVISIONS			
Ref. A Feedwater and High Pressure Lines These lines shall be fabricated from carbon steel, ASTM A-106 Gr. B. The minimum walls shall be as follows:			<div style="border: 1px solid black; height: 100px; margin: 0 auto; width: 100%;"></div>			
<table style="width: 100%; border-collapse: collapse;"> <tr> <td style="text-align: center; border-bottom: 1px solid black;"><u>Pipe Size</u></td> <td style="text-align: center; border-bottom: 1px solid black;"><u>Schedule</u></td> </tr> <tr> <td style="text-align: center;">2" thru 12"</td> <td style="text-align: center;">Schedule 80</td> </tr> </table>		<u>Pipe Size</u>		<u>Schedule</u>	2" thru 12"	Schedule 80
<u>Pipe Size</u>	<u>Schedule</u>					
2" thru 12"	Schedule 80					
All joints shall be butt welded for sizes 2-1/2" and over, and socket welded for 2" and under.						
Ref. B Reactor Service Lines These lines shall be fabricated from stainless steel, ASTM A-376 type 304. The minimum walls shall be as follows:						
<table style="width: 100%; border-collapse: collapse;"> <tr> <td style="text-align: center; border-bottom: 1px solid black;"><u>Pipe Size</u></td> <td style="text-align: center; border-bottom: 1px solid black;"><u>Schedule</u></td> </tr> <tr> <td style="text-align: center;">1" thru 4"</td> <td style="text-align: center;">Schedule 80</td> </tr> </table>		<u>Pipe Size</u>		<u>Schedule</u>	1" thru 4"	Schedule 80
<u>Pipe Size</u>	<u>Schedule</u>					
1" thru 4"	Schedule 80					
All joints shall be butt welded for sizes 2-1/2" and over, and socket welded for 2" and under.						
Ref. C Low Pressure Lines These lines shall be fabricated from carbon steel, ASTM A-53 or ASTM A-120. The minimum wall thickness shall be as follows:						
<table style="width: 100%; border-collapse: collapse;"> <tr> <td style="text-align: center; border-bottom: 1px solid black;"><u>Pipe Size</u></td> <td style="text-align: center; border-bottom: 1px solid black;"><u>Schedule</u></td> </tr> <tr> <td style="text-align: center;">1" thru 4"</td> <td style="text-align: center;">Schedule 40</td> </tr> </table>		<u>Pipe Size</u>	<u>Schedule</u>	1" thru 4"	Schedule 40	
<u>Pipe Size</u>	<u>Schedule</u>					
1" thru 4"	Schedule 40					
The joints shall be butt welded or flanged for sizes 2-1/2" and over; and socket welded for sizes 2" and under.						
Ref. D Main Steam Line This line shall be fabricated from alloy steel, ASTM 155 2-1/4% Cr, 1% Mo. The minimum wall thickness shall be as follows:						
<table style="width: 100%; border-collapse: collapse;"> <tr> <td style="text-align: center; border-bottom: 1px solid black;"><u>Pipe Size</u></td> <td style="text-align: center; border-bottom: 1px solid black;"><u>Schedule</u></td> </tr> <tr> <td style="text-align: center;">18"</td> <td style="text-align: center;">Schedule 100</td> </tr> </table>		<u>Pipe Size</u>	<u>Schedule</u>	18"	Schedule 100	
<u>Pipe Size</u>	<u>Schedule</u>					
18"	Schedule 100					
All joints shall be butt welded or flanged.						
PRINTS TO						
MADE BY P. E. Novak	APPROVALS <i>K. J. [Signature]</i>	DIV OR DEPT. APED	149A4222			
ISSUED <i>[Signature]</i> JAN. 3 '62	LOCATION San Jose, Calif.		CONT ON SHEET SH NO. 1			

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3.0 NUCLEAR STEAM SUPPLY SYSTEM

3.1 Reactor Vessel and Internals

3.1.1 Reactor Vessel

The reactor vessel, shown in Figure 3.1.1, is a right cylindrical pressure vessel with hemispherical heads. The design codes utilized are the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (Section I, Power Boilers, Section II Material Specifications, and Section VIII Unfired Pressure Vessels) 1959 Edition with latest addenda and code case rulings currently in force. Vessel specifications are given in Table 3.1.1.

The top head closure is a bolted, flanged closure using forty-six 5" diameter studs. The flange faces are clad with 304 SS by weld deposit as is the vessel interior. A double "O" ring seal is provided, and the area between the seals is monitored for detection of inner seal leakage.

The vessel is mounted on supports attached to the bottom head. Lateral stability for the top of the vessel is provided by four matched sets of guides attached to the surrounding structure. The guides are designed to allow axial and radial vessel expansion.

The feedwater sparger ring is made in two sections with one 8-inch line connected to each section through a thermal sleeve from the feedwater nozzle. The sparger section will be supported to the vessel wall in a manner to accommodate differential expansion.

The vessel-to-basin seal arrangement is a design feature that facilitates refueling. To permit flooding of the space above the reactor vessel, a basin seal skirt and flange are included with the vessel for mounting a seal plate. The seal plate is bolted to the surrounding dry well and forms a seal that permits filling the well above the vessel with water for refueling.

Safety valves are installed to protect the reactor vessel against excessive pressure build up. These valves are located in the steam line instead of the vessel dome to assure cooling for the fast core in the event of vessel over pressure. There are safety valves with a higher pressure setting located in the steam by-pass line, which would be used in the event of vessel overpressure during shutdown when the superheat core is flooded. The criteria for selection of safety valves is that the reactor pressure, with all valves open, must not exceed design pressure by more than 6 percent in accordance with the A.S.M.E. Code, in the event that the reactor, when operating at rated power, becomes completely isolated and fails to scram. Each of the four safety valves in the steam line have an A.S.M.E. rating (90 percent of the actual flow) of 291,000 #/hr, so that the total flow capacity exceeds the minimum ASME requirement of 150 percent of normal rated reactor steam flow.

TABLE 3.1.1

VESSEL SPECIFICATIONS

1. Straight Cylindrical Sheel (plate) With Nozzles

Thickness of base material - - - - -	4-3/4"
Thickness of cladding - - - - -	1/4"
Thickness base material plus cladding for an inside diameter of 11' - - - - -	5"

2. Hemispherical Plate Portions of Top Head

Thickness of base material - - - - -	4-3/4"
Thickness of cladding - - - - -	1/4"
Thickness of base material plus cladding - - - - -	5"
Inside radius bottom head - - - - -	5'6"
Inside radius top head - - - - -	5'6"

3. Pressure and Temperature

- (a) Design pressure 1250 psig
- (b) Design temperature 575°F
- (c) Working pressure 1,000 psig
- (d) Working temperature 545°F

4. Allowable Stresses of Vessel Material at Design Temperature

(Source ASME Boiler and Pressure Vessel Code, 1959 Edition, latest addenda).

Plate: SA302 Grade B Firebox Allowable Stress at 575°F - 20,000 psi

Forgings: SA336 Grade F-1 Allowable Stress at 575°F - 17,500 psi

Boltings: SA193 Grade B14 Allowable Stress at 575°F - 20,000 psi

Stainless Steel Fittings:	Pipe SA312 Type 304
	Forgings SA182 Grade F 304
	Allowable Stress at 575°F - 11,200 psi

5. Welds

Properly made welds inspected and controlled, according to Section I Power Boilers of the ASME Boiler and Pressure Vessel Code using approved material have an allowable stress at 575°F equal to 95% of the allowable stress of the material joined by the weld material.

6. Stress Limit of Material at Design Temperatures

SA302 Grade B Fire Box

Yield Strength (.2% offset)	35,000 psi min.
Tensile Strength	80,000 psi min.

6. (contd.)

SA336 Grade F1

Yield Strength (.2% offset)	32,000 psi min.
Tensile Strength	70,000 psi min.

SA193 Grade B14

Yield Strength (.2% offset)	100,000 psi min.
Tensile Strength	120,000 psi min.

SA312 Type 304

Yield Strength (.2% offset)	18,000 psi min.
Tensile Strength	58,000 psi min.

SA182 Grade F. 304

Yield Strength (.2% offset)	18,000 psi min.
Tensile Strength	58,000 psi min.

Yield and tensile strength at the design temperature are reported instead of creep stress, because creep stress data for most materials at this low temperature are not available.

7. Nozzles

Quantity	Size	Function	Location
1	2"	Vent	Top Head
20	3"	Top Mtd. Cont. Rod Drive	Top Head
14	1"	Flux Monitor	10 Bottom & 4 Top Head
5	4"	Instrument	Cylinder
1	2-1/2"	Emergency Cooling	
4	2"	Liquid Level & Press	
1	4"	Shutdown Cooling	
2	8"	Feedwater	
2	3"	Poison & Core Spray	
1	4"	Steam Bypass	Cylinder
1	2-1/2"	Clean-up Outlet	Bottom Head
1	16" I.D.	Steam Outlet	
44	4"	Control Rod Drive	
1	1-1/2"	Drain Nozzle	Bottom Head
2	20"	Recirculating Outlet	Cylinder
2	18"	Recirculating Inlet	Bottom Head

3.1.2 Coolant Flow Path

Water enters the boiling core through hollow nose pieces at the bottom of the fuel assemblies. It passes upward inside the channels surrounding each fuel assembly where it is heated, and steam is generated. The mixture of steam and water leaves the fuel channels at the top and enters the chimney, which provides the additional head necessary for the natural circulation flow of coolant.

The initial separation of steam from the water is accomplished by gravity at the free surface of the water just above the chimney. The steam then passes through a mesh type dryer where it is dried to about 99.8% quality. Dry steam passes down through the offset duct to the fast spectrum superheat core. The superheated steam from the superheat section passes through the shielding blocks below the core and out through the nozzle in the bottom head.

The water removed from the steam passing through the mesh type dryer joins the water leaving the chimney section to flow down the annulus between the core shroud and the vessel wall. Feedwater is added to the system through thermally sleeved spargers in the annulus below the top of the chimney where it joins the downward flow of recirculation water to the lower end of the vessel. Here the water enters the boiling water section inlet plenum through cut-outs in the lower portion of the core shroud.

3.1.3 Reactor Vessel Internals

The boiling water section outer shroud is bolted to lugs welded to the vessel near the bottom. This cylindrical shroud isolates the core from the downcomer annulus and has a flange at the top which mates with the chimney flange. The chimney, which extends upward from the shroud to provide additional isolation of the steam-water flow from the downcomer annulus, includes a core guide structure at its lower end to provide lateral support for the top of each fuel assembly. Poison curtains are hung from the top guide and extend downward between the fuel assemblies. These poison strips are used to hold down excess core reactivity during initial operation of the core and are removed prior to loading the equilibrium core.

The individual fuel assemblies, which make up the boiling water section of the core, rest on small fuel support plates mounted on top of the control rod guide tubes. Each guide tube is supported from the bottom head of the vessel and bears the weight of the four surrounding fuel assemblies. Two fuel assemblies in each quadrant, used to round out the periphery of the core to a more circular cross section, are outside of the pattern of control rod guide tubes. These "extra" assemblies rest on fuel support plates mounted on top of the core support assembly.

The core support assembly, which rests on brackets inside the core shroud, provides lateral guidance at the top of each control rod guide tube, but leaves the guide tubes free for independent vertical movement. Control

rods occupy the space between every other row of fuel assemblies and are withdrawn into the guide tubes below the core. Each control rod in the thermal core section is latched to a control rod drive mounted within a thimble welded to the bottom head of the vessel. The control rod drives for the thermal buffer section are mounted on the vessel top head, but the control rod movement is the same as in the thermal core section, i.e. downward out of the core.

In-core flux monitors are provided to obtain accurate flux measurements within the core. These flux monitors for the boiling water section of the core are housed in thimbles that penetrate the vessel lower head and extend through the core support to the top guide. Since the thimbles are located in the space between fuel assemblies, they do not have to be disturbed during refueling.

Primary separation of water from the steam is by free surface or gravity. The steam then passes through the steam dryer which is supported from the vessel wall near the top of the vessel. The mesh type dryer occupies the cross section of the vessel except for the inlet section of the steam downcomer and is inclined toward the center of the vessel to promote drainage.

The steam downcomer (offset duct) is rectangular in cross section and has large offsets along its length to provide water shielding directly above the superheat core region. The total flux at the vessel top head directly over the offset duct is calculated to be less than 10^6 fast neutrons/cm²-sec. The offset duct assembly is supported on a flange above the superheat core. This flange contains a double seal with a leak detection bleed-off between seals to allow monitoring for leakage and removal of a limited amount of seal leakage. The steam dryer provides lateral support for the top of the offset duct but permits unrestrained vertical movement to accommodate differential expansion.

The guide pipes for the fast core control bundle shafts pass through the offsets in the steam downcomer. They are welded to the offsets, as well as the lower flange, to give added strength to the assembly and to prevent leakage of reactor water into the superheat core. Instrumentation leads for the fast core will pass through a similar pipe permanently attached to the offset duct assembly.

The superheat core and associated components are all supported from the vessel lower head by the superheat core shroud assembly and are entirely independent of the boiling core support. This assembly is square in cross section and extends upward through the center of the boiling core region to the steam downcomer flange area. The fast core shroud immediately surrounding the fast core has relatively thin walls, which are designed to flex inward due to the difference in pressure between the superheat core region and the boiler during operation. This region of the shroud is supported against collapse by the strength of the fast core fuel assemblies, and, since it makes contact with the fuel assemblies, it prevents relative movement of the fuel in the superheat region. There

will be clearance for fuel handling during reactor shutdown since the core will be at a lower temperature, and without the pressure differential the shroud will be at its normal position.

The lower support grid for the superheat core and thus the core weight is supported by the rigid walls of the lower shroud section. The shielding below the core consists of alternate layers of borated steel and carbon cased in steel. Spacers are used between these shield assemblies to provide a void space for flow redistribution, since the openings for steam flow in one assembly are offset from those in an adjacent assembly to prevent neutron streaming. To allow for differential expansion, these shield assemblies and spacers are not attached to the fast core shroud walls but are supported from a ledge near the bottom of the fast core shroud. The limit of 10^6 fast neutrons/cm²-sec at the bottom head is used in this shielding design.

The transition from the square shroud to circular steam pipe is made just above the steam outlet nozzle. This transition region and the outlet steam line has a thermal liner providing a stagnant steam layer to insulate the high temperature steam line from the reactor water.

3.1.4 Steam Outlet Pipe and Seal Arrangement

There are three concentric pipes extending below the pressure vessel: (1) the thermal liner previously mentioned, (2) the pipe separating the stagnant steam layer from the reactor water, and (3) the pipe which is a continuation of the vessel nozzle. These pipes and the associated seal arrangement are shown in Figure 3.1.2. The two inner pipes are attached to the superheater shroud at the top of the transition section and are free to expand downward through the seal assemblies below the vessel.

The center pipe acts as the thermal liner and is sealed from the stagnant steam space by the lower seal assembly. This seal assembly has two sealing areas, each made up of expanding rings in a groove in the outer housing which are spring loaded against the pipe. The monitored leakage gland between the seal areas is connected to the condenser through a pressure reducing valve which maintains the gland pressure slightly below the steam line pressure to prevent stagnant steam leakage into the superheated steam region. The stagnant steam space is connected directly to the reactor steam dome so that it needs no pressure or flow regulation.

The reactor water is sealed from the stagnant steam region by a second seal assembly located above the one just discussed. This seal is a similar design and has the leakage gland pressure maintained slightly below the stagnant steam pressure to assure no leakage of reactor water into the stagnant steam region. This seal, located well below the pressure vessel, is accessible and eliminates the necessity of exposing the vessel nozzle to the stagnant steam which may be above reactor water temperature.

An overall advantage of the two seal assembly arrangement is that it permits sealing the reactor water from the steam with little pressure difference across the seal and, as a result, little tendency for leakage. The lower seal assembly is the "pressure seal" since it is subjected to the pressure drop existing across the superheat core section.

A major advantage of this seal arrangement as compared to a seal welded in place is that the complete superheat core assembly can be removed for repair or replacement without personnel exposure to the activated vessel or internals.

3.2 Boiling Water Core

3.2.1 Fuel and Core, General

The boiling water core is made up of 128 fuel bundles in the thermal core and 60 in the thermal buffer region. This arrangement is shown in the plan view of Figure 3.1.1.

There are 36 bottom-mounted, bottom-entry control elements in the thermal core and 16 top-mounted, bottom-entry control elements in the thermal buffer all on an 11-inch pitch. With this lattice and core size, there are some control elements located at the edge of the core. The resulting fuel space outside these control elements is occupied by 28 dummy fuel elements. This results in the same guidance for the control elements in this region as is provided for those toward the center of the core and also provides space for adding more fuel in anticipation of future increases in power as discussed in section 14 of this report.

Control rod followers, with a blade width and thickness equal to the full dimensions of the control blades, are provided to improve neutron economy and power distribution. The control rod followers in the thermal core region extend $3/4$ of the length of the core whereas the followers in the thermal buffer region are full length so the control rod can be completely removed from the core. The full length follower is necessary since these rods are driven from above and the coupling is too large to pass into the control rod space between fuel assemblies.

The control rod guide tubes located below the core not only guide the control rods and support the weight of the surrounding fuel assemblies but also provide a way of controlling the amount of coolant passing up through the control rod passage.

The top and bottom grid structures are located in line with the spaces between fuel assemblies and therefore allow the control rods and the control rod guide tube and fuel support assemblies to be readily removed. The chimney, the lower portion of which is the top grid, carries the flow from the four fuel assemblies surrounding a control rod and thus induces more flow through the hot channel where the void fraction is the greatest.

Provision is made for inserting stainless steel sheets containing small amounts of boron in the water gaps between control rods to provide temporary reactivity control for the first core. These poison sheets are hung from the chimney and may be removed by first lifting the adjacent fuel assemblies out of the core.

3.2.1.1 Core and Arrangement

Since the philosophy of the MSSR design is to rely on existing BWR technology, the boiling water core design is in many respects patterned after natural circulation BWR's of comparable power which have been designed by the General Electric Company. Two parameters which differ somewhat are the overall average power density and the core length to diameter ratio. The basis for this core size is discussed in the following paragraphs.

Physics considerations for this concept established the minimum dimensions for the fast core cross section of 28 inches square which, when adding approximately 2 inches on each side for the fast buffer fuel, resulted in an overall superheat core cross section of 32 inches by 32 inches. Based on this size and allowing adequate space for the coupling control rods around the superheater, a boiling core control rod pitch of 11 in. was established.

Physics considerations also dictated the thermal buffer thickness of approximately 10 inches and the thermal core thickness of approximately 15 inches which results in the boiling water core fuel layout shown on the plan view of Figure 3.1.1.

The control rod locations shown on Figure 3.1.1 were determined by the following considerations: (1) there should be control elements centered in the buffer region, and (2) the control spacing should be uniform across the boiling core region.

A core length of 80 inches was chosen for the prototype, since it gives approximately the same ratio of thermal core length to fast core length as the 300 MWe reference design. This core length results in a relatively low average power density of 21.1 KW/liter and a thermal core length-to-diameter ratio of 0.86. It is recognized that this design is not optimum from the standpoint of economics. It is more in keeping with the stated objectives of providing information that could be extrapolated to the design of a large power reactor. Provided that the initial operation at 75 MWe is favorable, it is planned to raise the power level without major changes to demonstrate the potential capabilities of the concept. The low power density lends itself to this philosophy, so the 80-inch thermal core length is used even though it is not the optimum length.

3.2.1.2 Fuel Assemblies

The boiling water core fuel assembly, shown in Figure 3.2.1, has 36 fuel rods, each containing 0.511 inch diameter UO_2 pellets with an active length of 80 inches. Even though the fuel assemblies and lattice in this design are larger than existing BWR core designs, it was believed that 36 rods, slightly larger in diameter, would be satisfactory since this core will be operated at a lower power density.

The cladding used in this design is 13 mil, 0.537 inch O.D. stainless steel. Stainless steel is chosen as the cladding material in the proposed fuel design since it is considered to be statistically more reliable although, from test results obtained to date, there is good likelihood that zircaloy-2 will be developed to a point where an economic choice can be made between the two materials.

The fuel rods are enclosed in a 60 mil Zr-4 channel and are positioned by an upper and lower tieplate and 3 rod spacers as shown in Figure 3.2.1. The fuel rods are attached to the lower tie plate and are spring loaded at the upper tie plate to allow for differential expansion. The nose pieces for the thermal core and thermal buffer fuel assemblies will be formed differently so they cannot be inadvertently loaded in the wrong core region.

The initial enrichment in the thermal core and the upper 42 inches of the thermal buffer is 3%. The lower 38 inches of the thermal buffer region which surrounds the superheat core is loaded with a mixture of Al_2O_3 and enriched UO_2 . This unusual fuel loading in the thermal buffer is necessary because of the fast flux leakage from the superheat core. This is discussed in detail in section 3.5.2.2 of this report.

The water moderator-to-fuel material volume ratio is approximately 2.6 for this stainless steel cladding design. This selection is based on consideration of physics performance, fuel cycle economics, and fuel design limitations.

3.2.2 Heat Transfer and Fluid Flow

The size of the boiling water core is based primarily on the physics considerations discussed previously and as a result appears very conservative from the thermal-hydraulic standpoint. Some preliminary calculations have been made for this core and are discussed below along with criteria used for these calculations.

The main design goal for the cooling system is to achieve equilibrium between heat generation and heat removal under all conceivable modes of operation without ever exceeding permissible fuel and cladding temperatures. To do this, the following factors, are considered to arrive at hot channel performance and overall peaking factor. These factors, discussed in section 3.5.5 of this report, relate the local power generation to the average for the whole core.

Radial and Inter-Control Rod	1.25
Axial	1.5
Local	1.3
Margin	1.15
Overpower	1.2

These factors are used to establish the overall peaking factor which expresses the design allowance for these effects on fuel element heat flux and temperature.

$$\text{Overall peaking factor} = F_{\text{R\&I.C.R.}} \times F_{\text{AX}} \times F_{\text{LOC}} \times R_{\text{O.P.}} \times \text{Margin} = 3.36$$

The enthalpy rise in the "hot channel", and thus the steam quality and void fraction, are not affected by the axial or local peaking factors since these apply to each channel in the core.

$$\text{Overall hot channel factor} = F_{\text{R\&I.C.R.}} \times R_{\text{O.P.}} \times \text{Margin} = 1.73$$

Results of preliminary calculations based on these factors are as follows:

Core Power - Including 20% Over Power, MWt	189
Maximum Fuel Temp. - Based on $K = 1.3 \text{ B/Hr } ^\circ\text{ft}, ^\circ\text{F}$	3400
Avg. Core Exit Void, % by Volume	29
Max. Heat Flux, Btu/hr-ft^2	290,000
Minimum Burnout Ratio	2.85

The preliminary calculations indicate good coolant flow capability for this core operating as a natural circulation reactor even with very little chimney, since the core size and coolant flow area through the core are relatively large. Here, as in Section 3.2.1.1, it should be noted that the thermal hydraulic design of this reactor is conservative. The reasons for the conservativeness is the same as before; namely, it is desired to incorporate as much flexibility as possible to provide for future increase in power output.

Separation of steam from the steam-water mixture leaving the core is accomplished by gravity or free surface separation within the vessel. Final removal of water remaining in the steam is accompanied by a mesh type dryer located near the top of the vessel. This type of dryer provides

a 99.9% exit quality with an average steam entering velocity of approximately 1 ft/sec. and steam inlet quality of 92% or higher. Test results obtained by Ongman (1) for free surface separation of 1000 psig indicate that efficiencies of 92% are feasible for average steam leaving velocities less than 1.1 ft/sec. and a minimum height of 30 to 34 inches above the indicated water level. The present design is well within the above limits. The average steam leaving velocity is 0.75 ft/sec., and the vertical distance between the water level and the lowest point on the steam dryer is 4 ft.

Steam carryunder is also an important consideration in a natural circulation reactor, since by decreasing the subcooling, steam carryunder increases the void content in the core. This in turn lowers the burnout margin and the reactor flow rate. The effect on the reactor flow rate is two-fold. Not only is the driving head reduced by the decrease in downcomer density, but also the resistance to flow in the core increases because of the effect of the voids on the two-phase pressure drop. Since little quantitative information on carryunder is available, a carryunder allowance of 1% by weight was made in the thermal hydraulic calculation. This is consistent with current design practice for the downcomer velocity for this reactor, which is calculated to be 1.44 ft/sec. in the annulus around the chimneys. This area requires further study, however, since recent ANL data (2) indicate that the threshold velocity for the inception of carryunder can be as low as 1.0 to 1.3 ft/sec., and that carryunder increases markedly above this threshold velocity. It is believed that the conservative thermal hydraulic design of the present reactor provides sufficient margin to accommodate the higher downcomer velocities.

3.2.3 Power Capability of Core

Detailed thermal-hydraulic calculations have not been made for a higher power level than the design discussed. However, it is believed that the limit for natural circulation operation would be reached at a plant gross electrical output of about 100 MW due to steam separation difficulties. The thermal-hydraulic performance of the core should be satisfactory at this higher power level since, with the additional 28 fuel assemblies installed, the average core power density would be 24.5 KW/liter which is below that in present natural circulation BWR designs.

Operation at the 100 MWt level would correspond to a thermal power output of 210 MW in the boiler. This would result in a steam flow rate of 760,000 lbs/hr., and the steam velocity leaving the free water surface would approach 1 ft/sec. The limit of steam separation is reached at this power level since at higher velocities, excessive water carryover would occur.

- (1) H. D. Ongman. Final Report on "Free-Surface Separation at 1000 psig" GEAP-3465 (July, 1960).
- (2) M. Petrick. Steam Water Separation Test, "Nuclear Superheat Meeting No. 5", COO 264 (Sept. 1961).

Steam carryunder may also be a limiting factor at the higher power level. The limiting conditions for steam carryunder, however, are more difficult to establish, since the factors relating downcomer velocity and carryunder are not well understood as yet. The thermal hydraulic design of the boiler is conservative, so that some margin is available to increase downcomer velocities and thus increase carryunder; however, any changes in this direction should be made with discretion. In 75 MWe operation, the downcomer velocity is 1.44 ft/sec., whereas for 100 MWe operation, the velocity would be up around 1.9 ft/sec. This is high, and it would require further justification.

3.3 Superheat Core Section

3.3.1 Fuel and Core

The size of this core, 32 inches square in cross section by 30 inches long, is established by the physics considerations which are discussed in Section 3.5.2. The fast buffer thickness and the fuel assembly dimensions are also determined by physics requirements. The fuel assemblies, which are shown in Figure 3.3.1, are limited in size. This is to limit the reactivity effect of the high fuel loading resulting from the large fuel volume fractions and high enrichment. It will also facilitate partial reloading of the fast section.

The fast core cross section, shown in Figure 3.1.1, is made up of 60 fast core fuel assemblies, 32 fast buffer fuel assemblies, and 4 fueled control assemblies. Each fast core fuel assembly is made up of 52 fuel rods consisting of 0.378 inch diameter fuel pellets clad in Hastelloy-X tubes. The cladding has an O.D. of 0.438 inch and a wall thickness of 0.030 inch. Each rod is made up of three sections loaded as follows: (1) depleted UO_2 in lower 12 inches, (2) enriched UO_2 in center 30 inches, and (3) depleted UO_2 in upper 12 inches. Allowance is made for fission gas release by providing a 3 inch void space in each rod between the top of the upper reflector and the end plug.

The fuel rods are held together loosely at the top and bottom by tie plates that permit vertical and lateral differential expansion. The lower tie plate assembly is attached to the stainless steel fuel assembly channel, whereas the upper assembly is free to move a limited amount to accommodate differential thermal expansion between fuel rods and channel. The main function of the upper tie plate is to facilitate fuel fabrication by holding the fuel bundle together during assembly. Filler pieces consisting of tube segments are installed on the fuel channels at the periphery of the fuel bundles to provide flow blockage and to aid in positioning the peripheral fuel rods. Since the filler pieces are hollow (with ends blocked off), they can be used as channels for in-core and below-core instrumentation leads.

The spacing along the length of the rods is provided by spiral wire wraps. Every third rod has no wrap, since it is positioned by the six surrounding rods, which are arranged with alternating direction of spiral wrap. In this way, it is thought that pressure drop will be held to a minimum and transverse mixing will be enhanced since the direction of coolant flow channelling between any two rods will always be the same. This rod array is similar to the reference fuel design developed by Aerojet-General Nucleonics under the Army Gas-Cooled Reactor System Program (1). The pitch-to-diameter ratio (rod spacing) is lower in the present design and therefore will warrant a developmental program to demonstrate thermal hydraulic characteristics. This program is discussed in Section 15.2.5 of this report.

The fuel assembly nose piece fits loosely into the lower support grid to prevent the fuel bundle from toppling over during the core loading operation. The loose fit facilitates loading and permits differential thermal expansion. Orificing of the core for flow variations to match the radial power is accomplished by attaching an orifice plate to the fuel assembly handle.

The fueled control elements, Figure 3.4.7, are similar to the fast core fuel assembly in construction, but are slightly smaller in cross section. A total of 42 rods per bundle is used instead of 52 as in the normal fuel assemblies. The control fuel rods have 30 inches of depleted UO_2 above the enriched "active" fuel instead of 12 so that when they are lowered to their minimum reactive position, i.e., fuel section below the core active fuel, large void spaces do not exist in the core.

The longer reflectors on the control fuel assemblies may require some adjustment to avoid restricting flow through the assemblies. If this adjustment cannot be made by orificing (enlarging to compensate for the longer flow path), it may be necessary to modify the design of the upper reflector rods to decrease its pressure drop.

The control element guide channel, Figure 3.4.8, has a flange at its lower end which rests in a recess on the lower grid and upon which the adjacent fuel assemblies are positioned. The guide channels are restrained from vertical movement by a hold-down grid between the top of the core and the lower section of the offset duct assembly. This arrangement provides clearance for the control elements, and does not alter the close packed fuel assembly condition in the core because the guide tube has the same outside dimensions as the fuel assembly channels. Clearance holes are located in the lower grid and the shielding assembly below the fast core to allow control element movement. There is no long nose piece on the control element guide tube since lateral movement is limited by the adjacent fuel assemblies. A relatively short tapered "lead-in" section is provided on the control

(1) AEC Research and Development Report IDO 28567

element to aid in starting it into the guide tube during refueling.

The 32 fast buffer fuel assemblies located around the fast core are similar in construction to the fast core fuel assembly but are different in size and fuel loading. The fuel and reflector material used here is natural UO_2 . The 28 assemblies located along the sides have 26 rods each, and the 4 corner assemblies have 42 rods each (see Figure 3.3.2).

3.3.2 Heat Transfer and Fluid Flow

Design of the superheater involves consideration of the interrelationships among heat transfer, pressure drop, fuel inventory, fabrication cost, and mechanical design problems. With the operating conditions (coolant pressure, temperature, and flow rate) and the overall size of the core established, the thermal hydraulic design consists mainly of selecting the fuel rod size and rod spacing. The selection made for this design is based on limiting fuel rod surface temperature and fuel pellet central temperature (1250°F and 4500°F respectively). The design based on these limits results in the highest fuel volume fraction (lowest fuel enrichment) largest diameter fuel rod, and minimum number of fuel rods. The lower enrichment is desirable because it gives a better Doppler coefficient. The minimum number of fuel rods and the larger rod diameter are desirable from the mechanical design and fabrication cost standpoint.

Although Hastelloy-X retains its strength up to about 1700°F , and has been used in reactors at this temperature,⁽¹⁾ it has not been proven under the proposed superheat operating conditions, so a conservative surface temperature limit of 1250°F is used. The major unknown involves erosion and corrosion in the superheated steam atmosphere. The fuel pellet central temperature limit, 4500°F , is based on a fuel conductivity of $1.3 \text{ Btu/hr-ft-}^\circ\text{F}$.

The following factors, discussed in Section 3.5.5, relating the local power generation to the average for the core are considered in the design so the limiting temperature will not be exceeded at the "hot spot" in the core.

Radial and inter-control rod flux peaking	1.5
Axial flux peaking	1.3
Margin	1.05
Overpower	1.2
Overall peaking factor	2.46

(1) The maximum surface temperature for Hastelloy-X clad in ML-1 is 1700° . EBOR clad is designed for 1510°F .

This design includes orificing of the inlet flow to provide a uniform temperature rise across each fuel channel. However, a uniform temperature rise may not be realized so allowance is made by using a flow factor of 0.87 which expresses the ratio of the flow rate in the hottest channel to the nominal flow rate. This factor accounts for such things as error in matching the flow and power distribution across the core, by-pass leakage between fuel channels, and mechanical effects such as assembly tolerances, fuel rod bowing and crud build-up.

The film coefficient, used in determining the fuel rod surface temperature, is calculated for parallel flow through the rod cluster, then modified by a factor of 1.2 to account for the effect of the spiral wire wrap. Allowance is made for circumferential variation in film coefficient by adding 20% of the maximum film drop to the calculated surface temperature at the "hot spot". These factors are based on results of heat and mass transfer studies conducted at Aerojet-General Nucleonics (1). However, since this design has closer rod spacing, i.e., lower pitch-to-diameter ratio, development work is essential in this area. This is discussed further in Section 15.2.5 of this report.

Some results of the heat transfer calculations are shown on Figure 3.3.3, where maximum fuel rod surface temperature is plotted as a function of fuel rod diameter for a number of pitch-to-diameter ratios. The design point indicated, $P/d = 1.05$ and rod diameter = 0.438 inch, is chosen since this is the combination which approaches the limits previously discussed for both fuel rod surface temperature and fuel pellet maximum temperature.

Pressure drop calculations, results of which are shown on Figure 3.3.4, are based on parallel flow through the core which neglects the blockage and channeling effect of the spiral wire wrap. A factor of 1.3 is applied to account for the effect of the wire wrap. This factor is based on the Aerojet - General Nucleonics test results for comparable Reynolds number. These curves show the reduction in pressure drop made possible by increasing the pitch-to-diameter ratio. When going to a larger P/d ratio, the fuel rod diameter must be decreased to provide more heat transfer surface and thereby remain below the surface temperature limit (See Figure 3.3.3) because the coolant velocity and film coefficient are lower for the increased P/d ratio.

As mentioned previously, the rod diameters and pitch-to-diameter ratios chosen for this design were determined by the fuel central and surface temperature limits. By permitting a lower central temperature, it would be possible to use smaller diameter rods with larger P/d ratios as shown in Figure 3.3.3. This was not done for the present design, since it was desired to simulate as closely as possible the physics and thermal hydraulic characteristics of the 300 MWe reference design.

(1) AEC Research and Development Report IDO 28567.

Thus, in this respect, more emphasis is placed on the role of the 75 MWe MSSR as a prototype rather than on economic power production. It remains to be seen whether heat transfer difficulties resulting from the low P/d ratio will force some relaxation of, for example, the central fuel temperature which would permit a higher P/d ratio. The final core design will represent a compromise of many parameters based upon test and development work and additional design analyses.

If development work on the proposed cladding, Hastelloy X, shows the feasibility of operating at temperatures higher than 1250°F, an additional degree of freedom will be available to improve the fuel design. Although, viewed optimistically, there appears to be no reason why Hastelloy cannot operate in temperatures higher than 1250°F in steam, very little data exist at the present time to warrant choosing a higher maximum temperature.

The design data for this core is summarized in Section 2.5, Plant Performance Data Summary.

3.4 Reactivity Control System

As compared with a conventional BWR, the reactivity control system provided for the present design reflects the more complex core arrangement of the Mixed Spectrum Superheater. Two sets of control drives controlling four core regions are specified. Bottom-mounted, locking piston drives are used for controlling the thermal core and for driving the coupling control rods between the fast and thermal cores. Top-mounted, twin-screw drives are used for controlling the fast core and thermal buffer. Both of these drives have been developed by General Electric for application in BWR's and are presently in use on operating reactors. Each of the drive systems is described in detail in GEAP-3590, "Steam Cooled Power Evaluation, Mixed Spectrum Superheater".

The locking-piston drives can be applied directly to the MSSR, since its application here is identical to standard General Electric BWR designs. The twin-screw drives will require some modification because, normally, the seals for the vessel penetration do not operate in a steam environment. Some development will be required as discussed in Section 15.0. As an alternative to the twin-screw drives, the magnetic jack drives used on PWR's have been considered. At the present time, it appears that either of these drives can be adapted to the MSSR.

3.4.1 Thermal Core Control System

In the thermal core, a typical control rod is centered in each group of four fuel assemblies. These control rods, of which there are 36 in thermal core, are bottom-mounted and bottom-entry. The rods are designed to move upward into the core to reduce reactivity. The

control rods (Dwg. No. 762D714, Figure No. 3.4.1) are cruciforms and contain a neutron-absorbing material which in this case is B_4C compacted in stainless steel tubes.

Power level of the reactor is increased by stepwise withdrawal of control rods in symmetrical pattern and power extraction from each section of the core is balanced by periodic adjustment of the control rod position pattern. Zircalloy followers of the same shape and width are bolted to the top ends of the control rods to improve core power distribution when the rods are partially or wholly withdrawn. The full stroke length is 80 inches.

The required lateral guidance for the control rod and follower is provided by hardened stainless steel rollers on the control rods, which bear against the cruciform control rod guide tubes, and on the followers which bear against the outside walls of the adjacent fuel channels. Extensive tests on a similarly guided control rod for Dresden, which was scrammed with a purposely defected water buffer for 200 consecutive times, have revealed, through subsequent inspection of the roller and guiding surfaces, that this type of guidance gives fully satisfactory operation.

Each control rod is individually positioned by a General Electric locking piston drive mounted in a thimble that penetrates the bottom head of the reactor vessel as shown in the reactor assembly drawing. Most of the drive is inside the thimble and is exposed to reactor pressure. The flange at the bottom of the drive is statically sealed to the reactor thimble using metal "O" rings and a circle of bolts. No rotary or linear shaft seals are required at the vessel penetration. Hydraulic lines are permanently welded to the thimble. Drilled passageways in thimble and drive flange provide the necessary connecting flow paths. Adequate pull space is provided below the reactor for drive removal. This arrangement provides direct access to any drive in the pattern. No extended wrenches or sequence involving removal of surrounding drives to provide access are required.

3.4.2 Coupling Control System

These eight control rods are located in the region between the fast and thermal buffers. Their purpose is to control the degree of nuclear coupling between the fast and thermal cores. At the present time, it is not planned to use these control rods for shim control. Provisions have been made to do so, if physics calculations now in progress indicate that some advantage would be gained.

The coupling control rods (or blades) consist of borated-steel, flat plates (Dwg. No. 762D719, Figure No. 3.4.2) driven from below. They are arranged around the fast core, just adjacent to and outside of the fast core shroud as shown in the layout drawing No. 104R674,

Figure 3.1.1. Offsets are required to couple the control blades to the drive shafts because the drives cannot be located directly under the blades due to the space taken up by the fast core lower shroud and the steam exit nozzle. A special guide tube Figure 3.4.3 is provided for these control rods since they are in the form of flat plates and have an offset to the drive shaft. The coupling control blades are individually driven by locking piston drives which are almost identical to the thermal core drives. The single difference is the stroke length, which for the coupling control is 30 inches rather than 80 inches. With this exception, the description of the drive system is identical to that of the thermal core drive system.

3.4.3 Fast Core Control System

In the fast core, reactivity is controlled by inserting or removing four fuel assemblies. These control fuel assemblies are similar in geometry to the standard fast core fuel assemblies but are smaller in cross section to allow space for the guide tubes. The control fuel assembly and guide tube are shown in Dwg. No. 762D722, Figure 3.4.4 and Dwg. No. 762D729, Figure 3.4.5. It is expected that the required lateral guidance in the guide tubes will be provided through the use of hardened steel rollers as specified for the thermal core control rods. For the fast core control bundles, however, no operating experience is available. Provision has been made in the Research and Development Program to investigate various coupling and guidance mechanisms. An extensive testing program is planned to provide assurance that operation will be satisfactory.

Each of the four control assemblies is individually driven by drives mounted on the top head of the vessel. The fuel assemblies are lifted up into the core to increase reactivity so that they scram down out of the core. The full stroke length is 30 inches. Indications of complete fuel assembly insertion or withdrawal, or of intermediate positions which each drive may set, is provided by a rod position indicating system.

3.4.4 Thermal Buffer Control System

The arrangement of the control rods in the thermal buffer is similar to that used for the thermal core, i.e., a control rod is centered in each group of four fuel assemblies. There are 16 control rods in the thermal buffer. At the present time, it is intended to use the buffer control rods primarily as a trimming control device to make fine adjustments in the power split between the fast and thermal cores. Preliminary physics calculations show that these control rods are not required for shutdown. Physics calculations now in progress indicate the strong possibility that the amount of reactivity control in this region can be reduced. The rods as shown, have been included

in the present design, since further work is necessary to demonstrate this. If later calculations confirm that the number of buffer controls can be reduced, significant gains in cost will be made.

The control rod geometry is similar to the control rods used in the thermal core. This can be seen in Dwg. No. 762D715, Figure 3.4.6. The thermal buffer rods, which are top-driven but bottom-entry, are likewise cruciform with zircalloy followers. The poison ($B_{4}C$) and follower section blades widths are slightly smaller (8.688 inches as compared to 9.50 inches) than the thermal core control blades. The reason for the smaller size is to provide additional space for the coupling control rods. It is possible that in later studies, it may prove to be advantageous to use rods of identical size in both thermal core and buffer. The stroke for the thermal buffer control rods is presently set at 74 inches, which is 6 inches shorter than the thermal core stroke.

With the exception of the stroke, the control rod drives used for the thermal buffer are identical to those used for the fast core controls.

3.5 Reactor Physics

3.5.1 General

The reactor physics analysis presents challenging problems in the design of the Mixed Spectrum Superheater. Much of the incentive provided by the concept is obtained by trading simplification of mechanical design and materials problems for increased nuclear complexity. Design of the Mixed Spectrum Superheater requires the simultaneous design of a fast-ceramic reactor and a boiling water reactor. The experience gained in General Electric's commercial boiling water reactor and the AEC-General Electric fast ceramic reactor programs and the analytical tools developed under these programs have been used liberally in the MSS reactor physics analysis.

3.5.2 Core Geometry

3.5.2.1 Size of Superheater and Boiler

The minimum size of a prototype Mixed Spectrum Superheater is limited by physics considerations concerning the fast core. As the fast core size is reduced the increased leakage requires the use of more highly enriched fuel. High leakage and high enrichments result in a harder neutron spectrum. These effects in turn reduce the magnitude of the negative Doppler coefficient which provides the prompt negative reactivity required to turn around an accidental excursion in the fast core. The increased enrichment and leakage also reduce the conversion ratio, and this lower conversion has an adverse effect on fuel costs. The 2.5 ft fast core used in the 75 MWe design has a Doppler coefficient about half of that of the reference 300 MW design of GEAP 3590. To reduce the core size further would make it difficult to extrapolate performance data to large size reactors and, more important, would compromise the safety of the prototype itself.

As can be seen from Figure 3.1.1, the height of the superheater core is less than half the height of the boiler. It is advantageous to make the fast core as spherical as possible. This reduces the fuel inventory, increases the internal conversion ratio, reduces the tendency towards large reactivity increases upon flooding, and increases the magnitude of the negative Doppler effect. From the standpoint of heat transfer and fluid flow the short core reduces the pressure drop and permits very close packing of the fuel rods thereby allowing the self-spacing wire-wrapped rod design illustrated in Figure 3.3.1. The close packing in turn benefits the physics by reducing leakage and, hence, improving inventory and conversion ratio. The wire spacers help to prevent bowing or vibration of the fuel rods and the potential hazard associated with these phenomena.

The dimensions of the boiling core have not been optimized but have been chosen to roughly conform to the geometric relationship between the fast and thermal core in the reference design. This requires operation of the boiler at a power density somewhat lower than is current BWR practice, which provides some safety margin, flexibility, and ability to increase the plant output.

3.5.2.2 Buffer Zones

The annular buffer zones located between the superheater and the boiler provide a transition from the fast spectrum and high power density of the superheater to the thermal spectrum and lower power density of the boiler. The thin steam-cooled buffer section (called the "fast buffer") is composed of fuel identical to that in the main part of the superheating core (called the "fast core") except for enrichment. It was assumed for the present design that natural uranium would be loaded initially into the fast buffer although enrichments up to 1.5% may be warranted. The fuel assembly geometry in the large moderated portion of the buffer (called the "thermal buffer") is identical to that in the main part of the boiler (called the "thermal core"). The first row of fuel assemblies in the thermal buffer contains fuel enriched in U-235 to 1.25% which is diluted with aluminum oxide (approx. 43% by vol) to decrease the power density while maintaining the proper conversion ratio. The second row of fuel assemblies in the thermal buffer contains 1.55% enriched fuel diluted with approximately 21% by volume aluminum oxide.

The fast buffer serves two functions. First it prevents most of the low energy neutrons diffusing from the boiler into the superheater from reaching the edge of the highly enriched fast core where they would cause a severe power spike; at the same time it is designed to allow sufficient transmission to increase the average importance of these neutrons sufficiently to enhance the control worth of the coupling control system (cf. Section 3.5.5.3). Second, it is designed to perform at high power density as a result of fissions caused by leakage of thermalized neutrons from the boiler. The reactivity worth of the coupling control system is dependent primarily on fission by neutrons leaking into the fast buffer. Fission of low energy neutrons from the boiler both in the fast buffer and near the edge of the fast core plays a significant role in determining the dynamic behavior of the reactor and is in larger measure responsible for the relatively long neutron lifetime ($\sim 3 \times 10^{-6}$ seconds) as compared to the lifetime of the fast reactor with no thermal coupling ($\sim 3 \times 10^{-7}$ seconds).

The function of the water-moderated thermal buffer is again two-fold. First its primary function is to isolate the very high power density fast core from the much lower power density thermal core so that desirable power distributions can be maintained in each region. Without the buffer the power at the edge of the boiling core would be excessive and the power away from the edge would drop off at a rapid rate. The thermal buffer prevents this by absorbing leakage neutrons from the superheater. In doing this it performs its second function which is to maintain a power density equal to that in the thermal core throughout the exposure of the thermal buffer fuel. To maintain this power density throughout life, it is necessary to choose carefully both the proper fissile material concentration and the ratio of fissile to fertile material. It is for this reason that different enrichments are used in the two rows of buffer fuel assemblies and that the fuel in the row nearest the high power density superheater is diluted with more aluminum oxide. The initial work showing that desirable power distributions can be maintained in the thermal buffer over long exposures (cf. Section 3.5.6.4) has been encouraging.

3.5.3 Fuel Inventory Requirements

In the superheating section, a fuel inventory of approximately 400 kilograms of U-235 is required in order to generate 24% of the entire reactor power. This member corresponds to a U-235 enrichment of 23.7% in a clean core in the hot operating condition with the fast core control assemblies two-thirds inserted. It is shown in the section on fuel cycle studies (cf. Table 3.5.5) that, for the equilibrium fuel cycle, the U-235 enrichment required in the fresh fuel is 26.28%. If Pu-239 is used instead of U-235 a fast core inventory of approximately 275 kg corresponding to a clean core enrichment of 16.1% is required.

3.5.4 Power Distribution

The radial power distribution through the center of the superheater and the adjacent lower part of the boiler is shown in Figure 3.5.1. This power distribution is the result of a four-group one-dimensional calculation. It closely represents the design power split of 24% of the total reactor power in the superheater. In determining the power split from a one-dimensional calculation the fact that the superheater is shorter than the boiler is taken into account. The power distribution in the upper part of the boiler above the superheater is shown in Figure 3.5.2.

A two-dimensional (R-Z) power distribution is shown in Figure 3.5.3. This calculation was made with the PDQ code. The superheater power represents only 15% of the total power in this case

as a result of the axial peaking of the power in the upper part of the boiler. The unreal assumption of a uniform axial void distribution made for the calculation accounts in large part for this axial power distortion; in actual operation the high void content of the upper part of the boiler will reduce the reactivity in this area and force the axial peak downward. The significant point is that it appears possible to design so that acceptable power levels and gradients are obtained in each of the many regions of the core.

Four-group one-dimensional neutron-flux distributions are shown in Figure 3.5.4. The calculation is the same as the one from which the power distribution in Figure 3.5.1 was obtained.

3.5.4.1 Methods Used to Obtain Power Distributions

All power distributions shown in this report are based on four-group calculations. The group structure is given in Table 3.5.1. The four-group cross sections in the fast core and the two buffer regions were condensed from a twenty-group cross section set for the neutron spectra obtained from a twenty-group one-dimensional calculation. The validity of this condensation was assured by close agreement in both critical enrichment and power distribution between twenty-group and four-group calculations. Methods used in conventional BWR technology were used to obtain cross sections in the thermal core. The epithermal cross sections for the thermal core were obtained from the fifty-four group GEM condensation code which is an adaption of the Bettis MUFT code. The thermal cross sections for the thermal core were obtained by averaging the cross sections over a Wilkins spectrum and accounting for spatial variations by use of P-3 spherical harmonics and two-dimensional FDQ calculations.

TABLE 3.5.1

Four-Group Cross Section Structure

<u>Group</u>	<u>Lower Energy</u>	<u>Lethargy</u>	
1	0.3 Mev	3.5	.939
2	9.1 Kev	7.0	.061
3	0.625 ev	16.6	0
4	Thermal	-	0

3.5.5 Peaking Factors

Overall peaking factors, or "hot spot" factors, must be determined for both the superheater and the boiler for the heat transfer analysis of the hot channels. The overall peaking factor is the convolution of several peaking factors which can be calculated individually. The overall peaking factor and the components for each section are listed in Table 3.5.2.

TABLE 3.5.2
Power Peaking Factors

	<u>Superheater</u>	<u>Boiler</u>
Radial and Inter-Control Rod	1.5	1.25
Axial	1.3	1.5
Local	1.0	1.3
Margin	1.05	1.15
Overpower	1.2	1.2
Overall Peaking Factor	2.46	3.36

The radial peaking factor is the peak-to-average power density ratio in the radial direction. In actual operation the radial power is "flattened" by relative control rod positioning (this flattening is not shown in Figures 3.5.1 - 3.5.8). Variation in power in the local area surrounding control rods contributes to an "inter-control rod" peaking factor. For the present analysis the radial and inter-control rod peaking factors are considered to be inseparable and are represented by a single factor. The radial and inter-control rod factors for both the fast core and the boiler were calculated for the power distribution in Figure 3.5.1 and are listed in Table 3.5.2. The value for the boiler includes the thermal buffer; the value for the superheater includes only the fast core. A slightly lower peaking factor was calculated for the upper part of the boiler, i.e. for the distribution shown in Figure 3.5.2. As a result of its annular shape it appears easier to shape the power of the MSS boiler than that of a standard BWR.

The axial peaking factor was calculated for the fast core from the two-dimensional power plot shown in Figure 3.5.3. The axial power distribution in the boiler is more complex because of the non-uniform steam void distribution. Therefore, the value for the axial peaking factor in the boiler used for present BWR designs of comparable size was assumed.

The local peaking factor accounts for lattice water gaps, self-shielding within the fuel rods, and local variations in fuel properties. In the fast core the lack of moderator and long mean free path result in negligible local power variation. For the boiler a local peaking factor comparable to that used for similar BWR lattices is used.

A margin peaking factor is specified to cover uncertainties in the above peaking factors. The boiler margin is the value assumed for present BWR design. The value assumed for the fast core is lower because there is no local factor and the radial and axial power distributions are more uniform than in the boiler. In addition, the clad of the superheater retains high strength far above its operating temperature. The 1250°C maximum surface temperature is chosen primarily because of unknowns in erosion and corrosion rates. Thus an overpower transient produces no dangerous effect in the superheater and the overpower peaking factor is in effect additional margin.

3.5.6 Control Systems

There are three control systems in the MSS: one in the boiler, one in the fast core, and one between the boiler and the superheater. The control systems in the fast and thermal cores are used during startup to bring the reactor to the hot operating condition. During normal power operation, they perform the dual functions of providing the reactivity compensation required for burnup and power shaping within each section and of controlling the power split between the superheater and the boiler. Shutdown reactivity in the boiler is provided by the thermal control system. Some shutdown reactivity in the superheater is provided by the fast core control system. Additional shutdown reactivity in the superheater is provided by the control sheets located between the boiler and the superheater. Since this is the only function served by this control system, the control sheets are completely withdrawn during normal operation.

In order to describe the functions of these control systems quantitatively it is convenient to specify three criticality factors. First is the effective static criticality factor, k , of the coupled system as a whole which is the ratio of the neutron population in the entire reactor between two successive generations.

It is the eigenvalue obtained from the time independent diffusion equation. Second the criticality factor, k_{eth} , is the multiplication factor of the infinite thermal core.* Third an effective criticality factor for the superheater, k_f , is defined as the criticality factor of the fast core, fast buffer, and thermal buffer without the presence of the thermal core. Both the fast and thermal buffers are included in the calculation of k_f since these two regions strongly effect the spectrum of and the net leakage from the fast core. The infinite multiplication factor of the fast core is not used for the present control study since leakage significantly effects the fast core characteristics.

In the sections which follow, the reactivity requirements in the superheater (in terms of k_f) and in the thermal core (in terms of k_{eth}) and the reactivity control available in each control system are examined. Next the effectiveness of the control systems in maintaining the desired power split is demonstrated.

3.5.6.1 Fast Core Control

The fast core control system consists of four movable fuel assemblies, as described in Section 3.4.3. Use of movable fuel instead of boron rods for fast control offers the advantage of better neutron economy and higher power density. In addition, an accident which blows a control element completely out of the core will result in a net loss of reactivity. Disadvantages are that the fuel assemblies are more massive than boron rods and that the reactivity worth of the fuel control assemblies varies with burnup more than for boron rods.

The reactivity requirements for the superheater for an equilibrium fuel cycle are estimated in Table 3.5.3. The burnup reactivity is based on the equilibrium burnup calculations for 50,000 MWD/t exposure discussed in Section 3.5.7.1. The reactivity control available in both the fast core control system and the shutdown control sheets between the superheater and boiler is also shown in Table 3.5.3.

The reactivity effect of the control assemblies decreases during the burnup cycle. However, this variation affects only the shutdown reactivity. Regardless of the burnup history of the control assemblies, the same 1.6% operating reactivity for

* This definition for the values of k_{eth} calculated and reported here is not rigorously correct since the energy spectrum of the finite thermal core was used for the calculations. Though this difference must be considered for detailed design, its significance to the control considerations discussed in the present conceptual design is negligible.

burnup and power shaping quoted in Table 3.5.3 will always be loaded into the core. The distance to which each fuel assembly is inserted to start the core up again depends upon its burnup history, but not the reactivity margin available for the burnup cycle. Since the control assemblies do burn, they constitute an integral part of the burnup cycle and must be replaced periodically (cf Section 3.5.7.1).

3.5.6.2 Thermal Core Control

The thermal core is controlled by means of B_4C incruciform control rods, as described in Section 3.4.1. In Figure 3.1.1 these rods are shown located throughout both the thermal core and thermal buffer. Location of control rods in the thermal buffer may be unnecessary since the multiplication factor of this buffer is substantially less than unity both during normal operation and in the cold condition. Since the thermal buffer is "driven" by the superheater, i.e., is dependent upon fast core neutrons as a source, changes in the thermal buffer multiplication factor during burnup are relatively unimportant.

The reactivity requirements for the initial and equilibrium thermal cores are estimated in Table 3.5.4. The equilibrium burnup reactivity is based on the equilibrium burnup calculations for 15,000 MWD/t exposure discussed in Section 3.5.7.5. Initial core burnup calculations were not made during the present study. The initial-core burnup reactivity reported in Table 3.5.4 is an estimate based on comparison with conventional BWR calculations for initial cores which attain an exposure 3500 MWD/t less than the equilibrium exposure. The other reactivities in Table 3.5.4 are also consistent with current boiling water reactor technology for similar size reactors.

The reactivity control available in the thermal control system is also listed in Table 3.5.4.

3.5.6.3 Coupling Control System

The coupling control system, also described in Section 3.4.2, is used solely for shutdown of the superheater. A mechanism is thus provided completely outside the central superheater pipe and with rod drives at opposite ends of the reactor from the fast core control rod drives for scramming the superheater.

It is planned to design this system to control slightly more than 2% in reactivity, the value quoted in Table 3.5.3, so that these controls will be able to shut down the fast core even if the fast core controls are in their most reactive positions.

Further effort is needed, however, since the present design provides a calculated effect of 1.2%. In the reference design of GEAP 3950 a 3% effect was obtained and it is believed that with careful redesign of the buffer the coupling control strength can be increased on the prototype without significantly effecting other areas of performance.

3.5.6.4 Power Split Between Superheater and Boiler

The power split between the superheater and the boiler depends upon the relative reactivity of the two sections and the degree of neutron coupling between them. The ability of the control system to alter the power split in a prescribed manner is discussed below. The ability of the control system to maintain the design power split during fuel burnup is implicit in the requirement that sufficient reactivity be controlled by the system to compensate for changes in reactivity resulting from burnup.

To study the ability of the control system to alter the power split, power distributions were calculated for several critical reactors differing only in superheater and boiler control rod positions. The results are shown as cases 1-3 in Table 3.5.5. The power distributions for these three cases are compared in Figure 3.5.5. Since each fast core control assembly is worth 1.1% in reactivity and the height of the fast core is 30 inches, the 0.4% fast core reactivity change necessary for the 15% change in power split observed in case 3 (cf. 24% of the power in the fast core in case 1 with 28% in case 3) is accomplished by a single fast core control element movement of ~1 inch. This relatively large alteration in the power distribution due to such a small change in rod position suggests the desirability of a finer shim control rod than exists in the present design. The -1% reactivity change required for the thermal core in Case 3 corresponds to inserting all thermal control rods about 4 inches.

During shutdown, the fast core is first shut down by withdrawal of the fast core control assemblies and insertion of the shutdown control sheets between the superheater and boiler (cf. Section 8.6.4). The superheater cladding temperature must be reduced sufficiently after scramming to allow flooding of the fast core. The thermal core must remain at power (though not at full power) to provide sufficient steam to cool the cladding. Calculations were made of the power distribution and the reactivity immediately following a superheater scram (i.e. fast core control assemblies withdrawn, shutdown control sheets inserted, thermal core control rod positions unchanged) and of the power distribution following the scram after the thermal core was returned to critical by the addition of 4%

reactivity. The results of these calculations are listed as cases 4 and 5 respectively in Table 3.5.5. The power distributions before and after these changes are shown in Figure 3.5.6. The small amount of power produced in the superheater under these conditions provides the rationale for the shutdown procedure described in Section 8.6.4.

TABLE 3.5.3
SUPERHEAT REACTIVITY CONTROL

<u>Reactivity Requirements</u>	
Cold unflooded to hot unflooded	.005
Burnup	.013
Power split and burnup margin	.003
Shutdown margin	.043
Total Δk	<u>.064</u>
<u>Control Available</u>	
Fast core control system	.044
Shutdown control system	.020

TABLE 3.5.4
THERMAL CORE REACTIVITY CONTROL

<u>Control Requirements</u>	<u>Initial Core</u>	<u>Equilibrium</u>
Cold to hot	.06	.04
Xenon and Samarium	.04	.04
Burnup	.13	.06
Power Shape and burnup margin	.02	.02
Shutdown margin	.05	.05
Total Δk	<u>.30</u>	<u>.21</u>
<u>Control Available</u>		
Control elements	.21	.21
Temporary or burnable poisons	.09	0

TABLE 3.5.5

Case	Control System Conditions	Reactivity Change		k	Fraction Power	
		Superheater	Boiler		Superheater	Boiler
		$\frac{\Delta k}{k} +$ k_+	$\frac{\Delta k_{\infty th}}{k_{\infty th}}$			
1	Normal operating	-	-	1.00	.240	.760
2	Fast control assemblies insert- ed, thermal control rods inserted	+0.005%	-0.095%	1.00	.245	.755
3	"	+0.04%	-1.06%	1.00	.280	.720
4	Fast control ele- ments withdrawn, shutdown control sheets completely inserted		-	0.97	.07	.93
5	Fast control ele- ments withdrawn, shutdown control sheets completely inserted, thermal control rods with- drawn ~ 12 in.		+4.0	1.00	.03	.97

3.5.7 Fuel Cycle

The fuel cycle for each of six regions in the reactor was analyzed. Concentrations of the fuel isotopes were calculated as a function of fuel exposure, and reactivity requirements to compensate for burnup were determined. The six different fuel regions are: fast core, fast buffer, fast reflector, inner and outer thermal buffers, and thermal core.

3.5.7.1 Fast Core

Total fuel exposure in the fast core was assumed to be 50,000 MWD/t. The reactor is shut down for refueling of both the fast and thermal cores at 14 month intervals. For the equilibrium fuel cycle one-sixth of the fast core fuel is replaced at

each refueling shutdown. One of the four fuel control assemblies is replaced during two out of every three refueling shutdowns.

Initial and discharge fuel compositions for the equilibrium cycle for a U-235 fueled fast core are given in Table 3.5.6. Values are shown for a plutonium-fueled fast core in Table 3.5.7.

The startup fuel cycle, that is before the equilibrium cycle is achieved, has not been analyzed. It appears that, unlike the thermal core, the limitations of fuel exposure will be governed by fuel and clad integrity rather than reactivity. If this is the case, fuel exposures for the initial cycle should be comparable to the equilibrium cycle.

For the equilibrium cycle the reactivity swing between fuel loadings, assuming uniform burnup, is 1.0%. Allowances for 0.3% reactivity penalty for nonuniform burnup and for an additional 0.3% for power shaping and burnup uncertainties were provided in the fuel cycle calculations.

3.5.7.2 Fast Buffer

The fast buffer is loaded in a single batch and allowed to burn for as long as fuel and clad integrity will allow. Total exposure is independent of power distribution or reactivity considerations. For the present economic evaluations an exposure of 50,000 MWD/t was assumed, although compositions are reported also for 100,000 MWD/t.

The initial and discharge compositions at both 50,000 and 100,000 MWD/t are given in Table 3.5.6. The concentrations are averaged over the fast buffer thickness. A large variation in the composition of the irradiated fuel exists across the buffer as a result of the extreme variation in neutron spectrum. This spatial variation was accounted for in the fuel cycle calculations by dividing the fast buffer into four half-inch thick sub-regions and calculating burnup in each sub-region. Most significant is the variation of the principal fissile isotope, Pu-239, the concentration of which is nearly 10 times as high in the half inch of buffer adjacent to the fast core where the thermal flux is low as in the half inch adjacent to the boiler where the thermal flux is high.

This variation in enrichment raises the interesting requirement that the fuel assemblies must be designed such that they can fit into the fast buffer in but one orientation. Otherwise, were an assembly to be removed temporarily for some reason, say for inspection, and turned around 180° before being replaced, a

prohibitive power spike would occur adjacent the boiler.

The power distribution in the fast buffer as a function of fuel exposure out to 100,000 MWD/t is shown in Figure 3.5.7. The discontinuities are not real but result from the mathematical device of calculating spatially varying burnup in four discrete sub-regions rather than continuously across the buffer. Starting with natural uranium the power nearly achieves its equilibrium distribution after 20,000 MWD/t. However, the power during the early exposure is so low that future design effort will be oriented toward achieving the equilibrium power distribution as early as possible by enriching slightly the fast buffer fuel.

As indicated above, the fast buffer of the 75 MWe MSS is not optimized. The calculated results do, however, show the characteristics of the buffer during life and show the direction to proceed to optimize it. For the present fast buffer design a period of eighteen years would be required to achieve an average fuel exposure of 50,000 MWD/t. This time could be reduced significantly (or the average exposure could be increased) by decreasing the fast buffer thickness. Both decreasing the thickness and increasing the enrichment are independently warranted by control considerations relevant to the shutdown control system between the superheater and boiler.

The effect of fast buffer burnup on the fast core reactivity is very small. During the first 14 month period (i.e., the period between refueling shutdowns) when the fast buffer power distribution was undergoing its most significant change, the fast core criticality factor changed by only 0.08%. The change was negligible after the equilibrium power distribution was achieved.

3.5.7.3 Fast Reflector

The fast reflector is exactly like the fast core except for enrichment. Depleted uranium is used in the fast reflector. The reflector extends one foot above and one foot below the fast core.

Since the reflector and fast core fuel form an integral assembly, the reflector fuel will be removed when the fast core fuel in the assembly receives an exposure of 50,000 MWD/t. The discharge composition of the fast reflector fuel, averaged over the entire reflector, is given in Table 3.5.6.

3.5.7.4 Thermal Buffer

The thermal buffer is loaded in a single batch and allowed to burn to an average of 30,000 MWD/t. The thermal buffer is

composed of two rows of fuel assemblies, each of slightly different composition to achieve simultaneously the required conversion ratio and power density in each row. It has not been proven yet whether the added power density achieved in the second row of elements offsets the economic penalty of the added fabrication costs incurred by having two types of buffer fuel; but the dual buffer concept is, nevertheless, illustrated by the present design.

The thermal buffer concentrations at average exposures of both 18,000 and 30,000 MWD/t are listed in Table 3.5.8 for both the inner and the outer row of thermal buffer assemblies. The exposures listed in Table 3.5.8 are averages over each row rather than the entire thermal buffer.

The concentrations vary significantly across each fuel assembly in the radial direction due to the rapidly changing neutron spectrum in the buffer. To account for this variation, the inner row of the thermal buffer was divided into three sub-regions and the outer row into two sub-regions (similar to the method used for the fast buffer), and burnup calculations were made for the average spectrum in each sub-region. The values reported in Table 3.5.8 were averaged over the sub-regions.

The thermal buffer burnup had a relatively small effect on the criticality factor of the reactor but a substantial effect on the power split. In the calculations, burnup was assumed to occur only in the thermal buffer; the superheater and thermal core concentrations were assumed to remain constant. The overall criticality factor increased by 0.2% during the first 3,000 MWD/t exposure of the thermal buffer and thereafter decreased to a total of - 0.8% at 30,000 MWD/t. The fraction of power in the superheater was 24% with the fresh buffer fuel, decreasing to 22% at 3,000 MWD/t, and thereafter increasing to 36% at 18,000 MWD/t and 45% at 30,000 MWD/t. The shift to 45% at 30,000 MWD/t may be too great to compensate with the present control system, and may require a refueling adjustment in the fast core. Actually, concurrent burnup in the fast core will effect the power shift as will refueling of the boiler. These effects are not taken into account in the calculation.

The power distribution in each sub-region of the thermal buffer is plotted as a function of burnup in Figure 3.5.8. The discontinuities between the two thermal buffer rows and between the thermal buffer and thermal core occur because of real step changes in fuel composition. The discontinuities within each thermal buffer row are not real but, as in the fast buffer (Fig. 3.5.7), result from the mathematical model. The power shapes at 18,000 and 30,000 MWD/t would be flatter in the actual

operating reactor than they appear in Figure 3.5.8; inadequacies of the mathematical model used distort the power distributions in the thermal buffer. Since the calculated fraction of power in the superheater is too high at 18,000 and 30,000 MWD/t the peak power in the buffer sub-region adjacent to the superheater appears higher than would be the case in actual operation. Similarly, the power in the thermal core and, hence, in the part of the thermal buffer adjacent to the thermal core appears lower than for actual operation.

3.5.7.5 Thermal Core

The equilibrium fuel cycle was calculated for a fuel exposure of 15,000 MWD/t. For the equilibrium cycle, one-fifth of the fuel is to be replaced at each refueling shutdown. Refueling shutdowns are scheduled every 14 months, coinciding with the fast core refueling shutdown schedule. Initial and discharge concentrations for the equilibrium fuel cycle are given in Table 3.5.9.

Fuel cycle data is estimated in Table 3.5.9 for the startup cycle. These estimates are based on comparisons of the startup and equilibrium cycle data for the Humbolt Bay Boiling Water Reactor rather than on calculations performed during the present study. Attainment of 11,500 MWD/t in the startup cycle requires the use of temporary poisons, such as the poison curtains to be used in the Humboldt Bay Reactor. These poison curtains would partially burn during the first core lifetime thus, continually adding reactivity. Near the end of the exposure they would be removed entirely to provide additional reactivity.

Several reactivity penalties were incorporated into the equilibrium fuel cycle calculation in order to present a sufficiently conservative result. A nonuniform burnup penalty of 3.1% was calculated to compensate for the large axial peak-to-average ratio. A 2.0% reactivity margin was assumed for burnup uncertainties other than those listed below and for margin for shaping the power near the end of the fuel cycle. Penalties for specific uncertainties in the burnup calculations included 1.0% for the effect of nonuniform plutonium production, 0.5% for the effect of nonuniform void distribution on conversion ratio and higher isotope cross-sections, and 0.2% for the effect of higher isotopes not included in the calculations. Finally, a penalty of 2.2% was calculated for the leakage out the central superheater steam pipe. Thus, the total reactivity penalty used for the thermal core burnup calculations was 9.0%.

3.5.8 Reactivity Coefficients

3.5.8.1 Doppler Coefficient in the Superheater

A prompt mechanism for turning around an accidental nuclear excursion in the superheater is provided by the negative Doppler coefficient. The increased value of this coefficient for a fast ceramic reactor over that of a metallic-fueled fast reactor results from the relatively large low energy tail in the neutron spectrum of a fast ceramic reactor, i.e., in the 1 - 10 keV energy range. It is in this energy range that the U-238 resonances contribute most strongly to the Doppler coefficient.

The methods described by Greebler and Hutchins * were used to calculate the U-238 contribution to the Doppler coefficient for the fast core of the 75 MWe MSS. A value of $-1.6 \times 10^{-5} \Delta k/k/^{\circ}C$ at room temperature was calculated for the U-238 contribution. The positive contribution by the U-235 has not been calculated although experience with comparable Pu-239 cores indicates that the magnitude of its contribution will be considerably less than half that of the U-238. This value is consistent with an earlier calculation of $-1.5 \times 10^{-5} \Delta k/k/^{\circ}C$ which was made for a Pu-239 fueled superheater similar in size and design to the 75 MWe and which included the positive contribution of the Pu-239.

The negative Doppler coefficient is roughly inversely proportional to the temperature in the fast ceramic reactor spectrum.

3.5.8.2 Doppler and Void Coefficients in the Boiler

The large negative Doppler coefficient in the boiler provides the prompt turn around mechanism for an excursion in the boiler. For a BWR of similar design, the Doppler coefficient is $-2.1 \times 10^{-5} \Delta k/k/^{\circ}C$ for the cold condition and $-1.6 \times 10^{-5} \Delta k/k/^{\circ}C$ for the hot operating condition.

Void formation is a slower mechanism which provides negative reactivity in an excursion and stabilizes the power level following operational or statistical changes during normal operation. The void coefficient is a strong function of the fuel exposure and the void fraction, always having a negative value of magnitude equal to or greater than about $1 \times 10^{-3} \Delta k/k/\text{percent void}$.

* P. Greebler and B. A. Hutchins, "The Doppler Effect in a Large Fast Oxide Reactor - Its Calculation and Significance for Reactor Safety", Proceedings of the Seminar on the Physics of Fast and Intermediate Reactors, Vienna (August, 1961).

TABLE 3.5.6
SUPERHEATER FUEL CYCLE DATA (75 MWe MSS)

U-235 FUELED FAST CORE

<u>Region</u>	<u>Fast Core</u>	<u>Fast Buffer</u>		<u>Fast Reflect- or</u>
Exposure (MWD/T)	50,000	50,000	100,000	
Time in Reactor (years)	7.3	18	36	7.3
Initial U-235 enrichment	26.28	0.71	0.71	0.30
Discharge U (% of initial U)	92.43	91.44	84.15	97.34
Final U-235 enrichment (% of discharge U)	21.84	0.19	0.17	0.28
Discharge Pu (% of initial U)	1.74	2.69	4.11	0.74
Breakdown of discharge Pu:				
Pu-239	1.70	1.45	2.23	0.73
Pu-240	0.04	0.90	1.23	0.01
Pu-241	-	0.17	0.29	-
Pu-242	-	0.17	0.36	-

TABLE 3.5.7

SUPERHEATER FUEL CYCLE DATA (75 MWe MSS)

Pu-239 FUELED FAST CORE

<u>Region</u>	<u>Fast Core</u>
Exposure (MWD/T)	50,000
Time in reactor (years)	7.3
Initial U (% of initial U + Pu)	82.35
Initial U-235 enrichment (% of initial U)	0.30
Initial Pu-239 enrichment (% of initial U + Pu)	17.65
Discharge U (% of initial U + Pu)	78.66
Final U-235 enrichment (% of discharge U)	0.22
Discharge Pu (% of initial U + Pu)	15.53
Breakdown of Discharge Pu	
Pu-239	14.51
Pu-240	0.99
Pu-241	0.03
Breakdown of Pu-239 history	
Pu-239 burned	5.95
Pu-239 produced	2.81

TABLE 3.5.8

THERMAL BUFFER FUEL CYCLE DATA (75 MW(e) MSS)

<u>Region</u>	<u>Thermal Buffer</u> <u>Inner Row</u>		<u>Thermal Buffer</u> <u>Outer Row</u>	
Exposure (MWD/t)	23,000	38,000	15,000	25,000
Time in Reactor (years)	4.7	8.1	4.7	8.1
Initial U-235 Enrichment	1.25	1.25	1.55	1.55
Discharge U (% of initial U)	96.63	94.34	97.72	96.51
Final U-235 Enrichment (% of discharge U)	0.14	0.02	0.51	0.24
Discharge Pu (% of initial U)	0.67	0.89	0.54	0.72
Breakdown of Discharge Pu:				
Pu-239	0.20	0.20	0.24	0.24
Pu-240	0.32	0.38	0.24	0.34
Pu-241	0.06	0.08	0.04	0.07
Pu-242	0.09	0.23	0.02	0.07

TABLE 3.5.9

THERMAL CORE FUEL CYCLE DATA (75 MW(e) MSS)

<u>Region</u>	<u>Thermal Core</u>	<u>Thermal Core</u>
Cycle	Startup	Equilibrium
Exposure (MWD/t)	11,500	15,000
Time Reactor (years)	4.7 (average)	5.7
Initial U-235 Enrichment	2.7	3.05
Discharge U (% of initial U)	98.2	97.59
Final U-235 Enrichment (% of discharge U)	1.6	1.57
Discharge Pu (% of initial U)	0.5	0.65
Breakdown of Discharge Pu:		
Pu-239		0.42
Pu-240		0.21
Pu-241		0.02

3.5.9 Reactivity Effects Accompanying the Removal of Superheater Fuel During Refueling

A condition peculiar to the refueling of a flooded fast superheater requires that thermal neutron absorbers be added to the channel walls of the superheater fuel assemblies.

The superheater is flooded during refueling in order to provide cooling and shielding. If thermal absorbers are not added to the channel walls, removal of the first few fuel assemblies from the flooded core actually results in the addition of a large positive reactivity. This situation is explained by the fact that the flooded superheater is highly undermoderated. Hence, most neutrons are absorbed at epithermal energies where their importance is a minimum, and few neutrons survive down to the highly important thermal energy region. When a fuel assembly is removed from the flooded core, however, the water hole left in place of the assembly provides a route for neutrons to by-pass capture in the epithermal region while slowing down to thermal energy. The number slowed down is sufficient to introduce a positive reactivity for each of the first few assemblies removed. The reactivity effect on removing the central assembly from the flooded core was calculated to be +4%.

This large positive reactivity can be prevented by placing a thermal absorber (e.g., gadolinium oxide or cadmium oxide) in the channel walls of each assembly in the amount of a few weight percent. Hence, neutrons which are thermalized in the water hole left by the removed fuel bundle would get captured rather than return to the fuel bearing region. For the case of thermally black channels removal of the central assembly from the flooded core was calculated to reduce the reactivity by 5%. It is presently planned to include enough thermal poison to produce a slightly negative effect upon removal of the first bundle.

It is noted that the presence of the thermal poison does not appreciably affect the core characteristics under normal operating conditions for which the thermal flux is negligible.

FIGURE 3.1.1

FIGURE 3.1/2 STEAM LINE SEAL ARRANGEMENT

- (G1) THERMAL CORE
- (G2) THERMAL BUFFER

SECTION A-A

SECTION B-B

SECTION C-C

FIGURE 3.2.1

REVISIONS	PRINTS TO

FIGURE 3.3.2

FIGURE 3.3.3 — MAXIMUM SURFACE TEMPERATURE
VS. FUEL ROD DIAMETER

FIGURE 3.3.4 – PRESSURE DROP VS. ROD DIAMETER

[illegible]

FIGURE 3.4.1

FIGURE 3.4.2

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9

GENERAL ELECTRIC

762D728

UNLESS OTHERWISE SPECIFIED USE THE FOLLOWING—				
APPLIED PRACTICES	SURFACES	TOLERANCES ON DIMENSIONS		
		FRACTIONS	DECIMALS	ANGLES
✓	✓	+	+	+

TITLE ASM
GUIDE TUBE
FIRST MADE FOR MCAP 3004B

COUPLING
CONTROL ROD

A

B

C

D

E

F

G

PP-412-P (6-61)

PRINTED IN U.S.A.

2

3

4

5

6

7

8

9

10

71

FIGURE 3.4.3

FIGURE 3.4.4

FIGURE 3.4.6

FIGURE 3.5.1 — RADIAL POWER DISTRIBUTION — 75 MWe MSS.

FIGURE 3.5.2 - RADIAL POWER DISTRIBUTION IN THE PART OF THE BOILER ABOVE THE SUPERHEATER.

FIGURE 3.5.3 - TWO-DIMENSIONAL (R-Z) POWER DISTRIBUTION-
75 MWe MSS. (UNNORMALIZED)

FIGURE 3.5.4 – RADIAL FOUR-GROUP NEUTRON FLUX DISTRIBUTIONS – 75 MWe MSS

FIGURE 3.5.5 - RADIAL POWER DISTRIBUTIONS FOR VARIOUS POWER SPLITS BETWEEN THE SUPERHEATER AND BOILER.

FIGURE 3.5.6 - RADIAL POWER DISTRIBUTION BEFORE AND AFTER SHUTDOWN OF THE SUPERHEATER.

RELATIVE POWER DENSITY

FIGURE 3.5.7 - RADIAL POWER DISTRIBUTION AS A FUNCTION OF BURN-UP IN THE FAST BUFFER.

FIGURE 3.5.8 - RADIAL POWER DISTRIBUTION AS A FUNCTION OF BURN-UP IN THE THERMAL BUFFER

4.0 REACTOR AUXILIARY SYSTEMS

4.1 Emergency Cooling System

An emergency cooling system is provided to remove decay heat from the reactor during scram and isolation of the reactor from the turbine plant. The cooling system, which is sized to continuously remove 6% of normal reactor power, consists of the following components: four high-capacity dump valves that can pass steam to an emergency low-pressure condensation tank located within the containment building; an emergency condenser located in the fuel storage pool, a de-superheater with its associated equipment, and; the necessary piping, valves, and controls. In addition, a by-pass line with a valve is provided between the top of the reactor vessel and the emergency condenser to remove decay heat in the event that the building isolation valve is closed and the fast core is flooded. The above components are shown on the P & ID of the Nuclear Steam Supply System in Figure 2.1.4.

The emergency condensing coil located in the fuel storage pool consists of two independent horizontal tube bundles for added safety. The differential elevation of approximately 18-1/2 feet between the cooling coils and the high-water level in the reactor provides for the return of the condensed drains to the reactor. Heat transferred to the fuel pool will normally be removed by the pool cooling system. If a power failure should prevent operation of the pool cooling system and addition of makeup water, the pool level will drop according to the amount of heat transferred. If it is assumed that 6% of the normal reactor power is transferred to the pool for a period of 8 hours under these conditions the level of the pool (20' wide by 27' long) will drop slightly more than one foot. Since the pool depth is 25', the shielding of the stored fuel will still be adequate. The steam generated would be exhausted through the plant ventilation ducts and discharged to the stacks.

The conditions under which the emergency cooling system must operate are:

1. Complete loss of normal station service power.
2. Generator trip-out, closure of turbine admission valves, and failure of by-pass valves to open.
3. Containment isolation valve closure.
4. Main condenser failure.

At the onset of the emergency condition an automatic signal scrams the reactor and simultaneously activates the emergency cooling system. The containment isolation valves begin to close instantly and the dump valves open directing steam to the emergency condensation tank. This results in initial steam flow rates of approximately 50% of rated. This high flow rate is required to assure that the superheater cladding temperature will not exceed tolerable limits. Calculations show that for this condition,

the cladding surface temperature will rise about 165°F before decreasing. After a time increment of approximately 30 seconds, only about 6% of rated flow is required to cool the superheater fuel elements. The dump valves are then closed so as not to remove an excess of primary system water. The emergency condenser and de-superheater system are then utilized to bring the system down to atmospheric pressure.

The dump valves, having flow capacity of approximately 50% of rated flow, are provided to cool the superheater fuel elements during the initial period of high reactor power decay and to dissipate the stored heat in the fuel. Four valves are provided on the primary steam lines within the enclosure, which pass steam to an emergency condensation tank. The valves are fast acting, closed to fully open within approximately one second, in order to prevent a substantial drop in steam flow through the superheater during the initial transient condition.

A de-superheater is provided in the steam lines to the emergency condenser to de-superheat the initial steam and thus prevent thermal shock to the condenser tubes. The de-superheater system consists of a high pressure accumulator, fast-acting valve in the coolant line from accumulator to de-superheater nozzles, and the de-superheater. When the emergency condenser steam line is initially opened, the de-superheater will be simultaneously brought into service. The de-superheater coolant lines and accumulator will be sized to remove all superheat from the initial steam flow to the condenser for three minutes of full flow operation. As pressure in the accumulator drops, flow will decrease and the steam flow to the emergency condenser will begin to pick up superheat. By the time the accumulator pressure drops to primary system pressure, the emergency condenser will be adequately warmed and capable of accepting superheated steam. The accumulator is connected to a pressurized (3,000 psi) air tank and compressor. Makeup water is provided by the control rod drive feed pump. Normally, de-superheating could continue on an indefinite basis.

4.2 Shutdown Cooling System

The primary function of the shutdown cooling system is to cool the reactor water down to about 120°F and to maintain this temperature during the refueling operation. The system may also be used to heat the reactor water to approximately 240°F during the first startup of the plant or with clean fuel where no decay heat is available.

The system consists of a closed loop serviced by two parallel cooling units. Each unit comprises a recirculating pump, a heat exchanger, and the necessary piping, valves and instrumentation.

The heat exchangers are designed for counterflow with reactor water on the tube side and service water on the shell side. The system components are designed for a maximum operating pressure of 150 psig. Pipelines up to the first shut-off valves, including the valves, are designed for an operating pressure of 1200 psig. Carbon steel piping and valves are used throughout. The tubes in the heat exchangers are made of copper-nickel alloy. All valves and components are welded to the pipe in the system.

The details of a normal shutdown are discussed in Section 8.0, under Plant Operation. The shutdown cooling system is turned on when the system pressure has been reduced to about 50 psig which is reached about three hours after reactor shutdown. The system is sized to absorb the entire reactor decay heat after this time. The total cooling capacity of the heat exchangers is about 1.0% of normal full-power operation. Approximately five to seven hours are required to cool the reactor system down to 120°F so that refueling can begin.

4.3 Reactor By-pass Cleanup System (Dwg. No. 762D725, Figure 4.3.1)

The reactor coolant is maintained at low solids concentration (< 0.5 ppm) by utilizing the reactor by-pass demineralizer system. Low solids are required for the following reasons:

1. To maintain good heat transfer in the core by reducing the deposition of solids on the fuel elements.
2. To reduce the secondary sources of beta and gamma radiation due to deposition of irradiated corrosion products.

The continuous cleanup of the reactor coolant is accomplished with a minimum of heat loss and no water loss from the cycle during normal operation.

During operation, reactor water under reactor pressure is passed to the tube side of a heat exchanger, exchanging heat with reactor feedwater, through a filter, a mixed bed demineralizer and to the condenser hotwell. During filter backwash or resin changing operations, the cleanup flow can be stored in the Clean Waste Tank. Flexibility is also provided by allowing the demineralizer effluent to be routed to a Waste Holdup Tank for sampling before disposition.

A pump is provided to circulate flow to the cleanup system during reactor startup or shutdown when the reactor pressure is low. Condensate is recirculated through the shell side of the heat exchanger and back through the feedwater recycle line to the main condenser. Feed to the reactor may be accomplished via the control rod drive feed system pumps, if the main feed pumps are not operating.

Chemical regeneration of spent resin is not normally carried out because of the high radioactivity of the impurities removed from the reactor coolant. Spent resins are sluiced from the demineralizer vessel to the rad waste system for disposal. Fresh resins are hydraulically transferred to the cleanup demineralizer from the resin storage tank in the Condensate Demineralizer System. During initial startup operation, used cleanup system resins may also be hydraulically transferred to this storage tank for dirt removal or regeneration and reuse.

4.4 Auxiliary Components Cooling

Two systems of component cooling water are furnished: the service water system and the circulating cooling water system.

The service water system employs river water for cooling components with very low or no radioactivity including the shutdown heat exchangers, fuel pool heat exchangers, reactor enclosure drain tank cooler, and the air conditioning equipment. Water from this system is monitored and discharged with the main condenser cooling water effluent. The design for this system assumes 65°F inlet temperature, and a 20°F temperature rise; on this basis approximately 2,000 GPM (maximum) of cooling water is required during refueling and 1,000 GPM (maximum) is required at full power.

The circulating cooling water system employs demineralized water, recirculated through a heat exchanger, to cool equipment which might contaminate the effluent with a high degree of radioactivity in event of a tube failure, or which might scale or plug with river water. Equipment cooled by this system includes the sample coolers, reactor cleanup demineralizer non-regenerative heat exchanger, pump oil coolers and bearings, and the reactor shield cooling coils. The design for this system assumes 90°F inlet temperature and a 30°F rise. On this basis a maximum of 500 GPM is required at full power. During shutdown 300 GPM will suffice.

4.5 Demineralized and Condensate Water Storage Systems

Condensate water is stored in two large tanks. One of these tanks is the condensate storage tank and the other is the demineralized water storage tank. The condensate storage tank is provided to store slightly contaminated demineralized water. This tank is connected to the condensate system so that it can supply makeup in the event of low water level in the hotwell and receive excess condensate from the downstream side of the condensate demineralizers when the hotwell level rises above normal. Makeup and reject to the condensate system is controlled automatically.

Normally, there is sufficient water in the condensate storage tank to deliver water by gravity plus condenser vacuum to the condenser. However, in the event there should be insufficient head and the water level in the hotwell should go below the low level control point, a second low level float switch operates to start one of the condensate transfer pumps.

Water is also supplied by gravity to the rod drive feed pump from the condensate storage tank for use as auxiliary feed. Water can be transferred from the demineralized water storage tank to the condensate storage tank, but the reverse transfer is not possible in order to avoid radioactive contamination of the demineralized water storage tank. The condensate storage tank receives previously contaminated water after proper treatment in the waste disposal system.

Overflow from the condensate storage tank discharges to the waste disposal system. The slightly contaminated demineralized water services are as follows:

1. Makeup.
2. Emergency condenser.
3. Resin sluice water.

4. Contaminated equipment flushing or filling.
5. Cooling water loop - radioactive equipment.
6. Fuel storage pool.
7. Any other demineralized water used where slight contamination does not add to radiation hazard.

The demineralized water storage tank is provided to store uncontaminated demineralized water. Water can be transferred from this tank to the condensate storage tank as required for steam cycle makeup or other use points supplied from the condensate storage tank. One demineralized water transfer pump is provided for this purpose. The makeup demineralizer is the only source discharging into the demineralized water storage tank. The makeup demineralizer is operated intermittently to keep this tank approximately full.

When demineralized water is necessary to replace the losses from the condensate and feedwater system, as indicated by decrease in level in the condensate storage tank, the demineralized water transfer pump is started manually. When the level indicator for the condensate storage tank (or its high level alarm) indicates that this tank has been filled, the pump is stopped. The overflow of the demineralized water tank discharges to the river since this water is no higher in radioactivity than that of the raw river water.

The clean demineralized water services are as follows:

1. Cooling water loop non-radioactive equipment.
2. Fuel cask and equipment decontamination stations.
3. Floor washdown where drains go to radioactive waste system (non-radioactive areas may use river water).
4. Demineralized water for laboratory.
5. Any other demineralized water uses where radioactive contamination is not present or desired.

The condensate storage tank has a level transmitter and indicator on the feedwater heater console. Conductivity of the water in this tank is measured and recorded on the heater panel. High and low level float switches are connected to the annunciator on the heater panel to alert the operator.

The demineralized water storage tank has a high level float switch connected to the annunciator on the heater panel.

4.6 Core Spray Cooling System

The function of the core spray system is to provide means of cooling the

reactor if the normal cooling capability is accidentally lost and to augment the cooling of the fast core during startup and shutdown. In addition, flooding of the superheating section is accomplished via the superheater portion of the spray system.

The core spray will be automatically actuated should low water level and low reactor pressure occur. The system can also be started manually from the control room. The system consists of two full capacity electrically driven pumps, automatic control valves, instrumentation, and the associated piping. Primary pump suction will be from the condensate storage tank with an alternate source piped from the station service water system. The alternate source can be used if the water in the condensate storage is exhausted. The water is pumped directly into the reactor vessel where it is distributed over the boiler and superheater cores by separate groups of spray nozzles. The system is rated at 350 gpm at a maximum pressure of 1250 psig. The pumps and control valves are located in the turbine building outside of the reactor enclosure so that they are accessible for routine maintenance during normal plant operation.

In the superheater portion of the spray system, an additional loop is provided to circulate saturated water at reactor pressure from the boiler to the superheating core. This arrangement can be seen on the NSSS P & ID, Figure 2.1.4. The suction end of the pump is connected to the line leading from the reactor to the shutdown cooling system. The circulated water is directed into the superheater portion of the spray cooling system.

The purpose of this loop is to provide a source of high-temperature water to cool the superheater fuel at conditions when injection of cold water would thermally shock the fuel. For example, during a normal shutdown, which is discussed in detail in Section 8, it is desirable to use spray cooling while the reactor is brought down from operating to atmospheric pressure. Since fuel temperatures will be of the order of 700°F when the spray is turned on, the use of high-temperature (550°F) water is necessary to prevent damage due to thermal shock. As discussed in Section 8, the successful utilization of this scheme must be established as a part of the proposed development program.

4.7 Liquid Poison System

If an equipment malfunction should prevent the control rods from shutting down the reactor, the Liquid Poison System is manually placed into operation by the reactor operator. A sodium pentaborate solution is used for this system and is injected through the poison sparger. The solution mixes with the water in the reactor vessel and causes the reactor to go subcritical because of the high neutron cross-section of the boron.

The system consists of a tank with a capacity of approximately 900 gallons in which a sodium pentaborate solution is stored under atmospheric conditions; two 1250 psi, 15 gpm positive displacement pumps installed in parallel; a 7.5 gpm recirculation pump; and the appropriate piping, valves and instrumentation. A recirculation system is provided to maintain the liquid poison system in operable condition at all times. If the

temperature should fall below 68°F, a heater is provided in the recirculation system to prevent precipitation of borate. Two remote operated valves in series are installed to prevent accidental injection of poison into the reactor and to provide a means of testing the injection system without actually introducing poison into the reactor. The valves are solenoid-operated and are energized by separate switches in the control room. Both switches must be energized to start the pump and open the discharge valves. Check valves placed downstream of the control valves prevent backflow of the reactor water, if the control valves are opened before the pump is put into operation.

The storage tank is provided with high and low level alarms and with low temperatures annunciated. The poison pump discharge pressure is transmitted and indicated in the control room.

4.8 Process Sampling System

Suitable equipment is required in the plant for continuous monitoring and/or intermittent sampling of radioactive and non-radioactive streams in order to supply information on process variables. Variables to be monitored include the following: water quality parameters of pH and conductivity for use in corrosion control; radiation levels of both radioactive and non-radioactive streams; and dissolved oxygen levels.

At critical process points, the above variables are continuously monitored. Similarly, in certain non-radioactive streams, which potentially may be contaminated with radioactive fluids, continuous radiation monitors are provided for the purpose of either preventing gross contamination of non-radioactive equipment or preventing significant radioactive material from reaching the environs. A detailed description of the process radiation monitoring is included in another section of this report. All continuously monitored process variables are recorded and/or annunciated in the main control room. In addition, sampling equipment is installed throughout the plant to provide batch and grab-sampling of solutions for laboratory or portable instrument analysis.

All sampling lines to the points of sampling are of stainless steel tubing or other materials suitable to the particular conditions encountered. In general, where the stream to be sampled is at high temperature, a cooler is provided. Where the stream is at high pressure, an orifice or pressure reducing valve is provided. All sampling lines with continuous monitors are piped to a closed drain with provisions for taking a grab sample. Adequate shielding is provided for sampling or monitoring equipment on highly radioactive streams.

4.9 Fuel Element Rupture Detection System

If a break should occur in the cladding of a fuel element, the plant may continue to operate as long as the quantity of fission products released to the atmosphere do not exceed the permissible emission rate. If the number of failures increases so that this emission rate is exceeded, it is necessary to locate and replace the failed assemblies. The fuel locating equipment is provided for this purpose.

To determine the general core location of ruptured assemblies in the thermal boiling core, a core flux-tilting procedure is used in conjunction with the off-gas monitoring system prior to shutdown of the reactor. A significant local depression of neutron flux by insertion of a control rod will reduce the operating heat flux of a ruptured assembly and thus reduce the release rate of fission products from a cladding break. This results in a reduction in off-gas activity continuously monitored by the off-gas monitoring system. Typically, the control rods in the north half of the reactor are inserted while withdrawing the control rods in the south half to maintain reactor power at whatever level is permitted by activity in the off-gas system and burnup level of the fuel in the core. The process is repeated using control rods in the east and west halves of the reactor. The quadrant where insertion of control rods reduces fission product activity is thus identified. Insertion of individual control rods within the quadrant further narrows the search area. Similarly, if the neutron flux is increased, fission product release rate will increase.

After the reactor is shut down, and the vessel head is removed, a channel plugging and sampling device known as a "sipper" is used to remove a representative water sample from an individual suspect fuel channel. The sipper is positioned with the fuel grapple to cap the top of a suspect fuel channel in the core. Continuous air flow into the cap establishes an air pocket above the fuel that prevents water sample contamination by cross-mixing from the surrounding fuel channels. After a short wait to allow for build-up of fission product iodine concentration, a water sample is withdrawn by vacuum to a sampling connection on the refueling platform through a flexible hose connected to the sipper. Fission product I-132 is separated chemically in the plant laboratory. The I-132 sample is counted with a scintillation detector and linear amplifier that can be biased at selected energy levels or with a single channel analyzer. The chemical separation is standard equipment for a nuclear plant chemical laboratory. The sample from a channel with defective fuel will have a higher count rate than samples from channels containing sound fuel.

The procedure described above is most effective during the 72 hours immediately following a plant shutdown. The small relative difference in sample activity between sound and defective fuel assemblies makes it difficult to identify a defective fuel assembly after this period.

The sipper, shown in Figure 4.9.1, is remotely controlled from a portable panel on the work platform above the reactor. Panel and sipper are connected by a set of flexible hoses. Service air and demineralized water supply lines are connected to the control panel. Vacuum is provided by a jet pump built into the panel.

The air cylinder, shown in Figure 4.9.1, is used to lift the fuel assembly being sampled out of the close fitting lateral support grid at the top of the core and up into the bubble cap for proper isolation from neighboring channels.

To detect defected fuel rods in the superheating section, sample collecting tubes are mounted in the grid support plate under each fuel assembly. Tubes are also located at the exits of the control fuel element guide channels. The tubes run along the base of the grid bars and are grouped into quadrants. Each set passes through a nozzle in the vessel wall and into a tube connection flange. Welded to the outside of each flange are

external connecting tubes, each of which leads to a common header. The four headers are individually monitored to identify the quadrant in which the defected fuel is located. Once the quadrant is identified, it is planned to use a sample probe similar to the sipper to locate the defected fuel bundle.

Leads from the four monitoring system go to a common header which then leads the samples to a condenser where entrained gases are separated. The undissolved gas mixes with the air carrier system and is passed through the monitor into the stack gas system. Condensate flows through a flow meter to a collecting tank and finally to the waste system.

Normally, it is planned to use the rupture detection system only after the monitoring system for off-gases indicates the presence of fission gases. When the ruptured fuel element has been identified, the headers, degassifiers, and sample gas lines are purged free of radioactive gases by the purging air system.

Since the fast core fuel rupture detection system can provide a continuous sampling of steam leaving the fast core, it offers a means of determining the performance of the fuel cladding. By passing part or all of the effluent from the detection system through a filtering system (e.g., stainless steel mesh) for long periods of time, the erosion and corrosion rates can be measured. This will provide an additional check on fuel performance.

4.10 Emergency Power Supply

Emergency power required for maintaining the nuclear steam supply system in a safe standby condition falls into the minimum power required immediately following the loss of the a-c auxiliary system and minimum power required after the station has been off for some seconds. A battery power supply and a 300 kw diesel engine driven generator are provided to furnish the necessary power to satisfy the requirements listed below.

4.10.1 Power Required Immediately Following Loss of Plant Auxiliary A-C System

The reactor safety system is designed to scram the reactor or make the reactor subcritical on loss of station auxiliary power. To assure that the plant is in a safe condition, the operator must be able to observe that the reactor has been made completely subcritical. Emergency power is provided to keep the following items operable at all times:

1. One neutron monitor chamber. (Provisions to transfer to another chamber in case normal chamber on this circuit is inoperable.)
2. Control rod position indication system.

3. Isolation valves in all lines needing isolation which do not close on loss of power.
4. The emergency cooling system--for example, power is provided to open valves to the emergency condenser.
5. The liquid poison system.
6. The emergency lighting in the control room to be sufficient for the operator to observe critical instruments also operated and built from the emergency supply.

This emergency power will be supplied from batteries, and in the case where a-c power is required for the neutron monitors and rod position indicators, etc., an inverter operated from the station battery supplies the a-c power requirements.

4.10.2 Additional Power Requirements

In addition to the battery power supply described above, an emergency power source in the form of a diesel generator will be available to provide power for extended periods following loss of normal station power. This will permit the operator to monitor important parameters in the reactor system to keep the reactor shut down. Also, power will be available to operate, for example, the core spray system, the cooling system which reduces pressure in the containment vessel, and area radiation monitors, if the maximum credible accident should occur.

4.11 Reactor Servicing System

4.11.1 Reactor Refueling System

4.11.1.1 Refueling Cycles

The Mixed Spectrum Superheating Reactor is designed to be refueled on a partial or batch schedule. Equipment is designed and the plant arranged to provide the maximum flexibility for the operator to meet the specific needs of the plant.

4.11.1.2 Refueling Equipment Description

The refueling isometric drawing, Figure 4.11.1, shows the general building and equipment arrangement provided for refueling. A well is provided above the reactor vessel so that the water level can be raised to an elevation matching the storage pool water level. A slot is provided on the wall between the reactor well and the storage pool for underwater transfer of spent fuel assemblies from the core to fuel storage racks in the pool, for decay storage prior to shipment for reprocessing. A slot gate and cover is provided to prevent leakage from the storage pool when the water has been drained from the reactor well.

A refueling platform rides on rails straddling the reactor well and the storage pool. A power operated fuel grapple, Figure 4.11.2, similar to the one used at Dresden is mounted on a monorail on the platform and extends through a long slot in the platform floor. One operator can perform the refueling operation. By use of the control panel mounted at the top of the grapple, he can move the platform, traverse the grapple and extend or retract the grapple head. He also controls the fuel grapple engaging mechanism from the panel. A small auxiliary hoist is provided for general lifting assistance. The operator can see the entire fuel handling operation through the slot in the refueling platform floor.

The reactor building crane also traverses the reactor and storage pool area and is used for handling heavy pieces of equipment, such as the shielding above the reactor vessel, and the reactor vessel head.

The fuel storage pool is arranged to provide enough fuel storage racks so that the entire core can be unloaded and still have additional space for a 20% batch of fuel undergoing decay prior to shipment for reprocessing. The fuel racks contain poison to prevent criticality of the compact storage arrangement. Storage space is also provided to store a full complement of control rods.

The highly enriched superheater fuel will be stored in separate racks with no more than four bundles grouped together in one slot. Each group will be spaced with a minimum of six inches between groups to prevent the inadvertent formation of a critical mass. It is estimated that 16 fuel bundles (assuming 3.5" square bundles) are required to form a critical mass in water.

The fuel storage pool is steel lined, 27' long x 20' wide x 25' deep. Approximately 16' of water over the fuel assemblies provides shielding to limit radiation exposure. The pool is sized to store the following:

- 8 thermal fuel storage racks (28 fuel assemblies per rack)
- 20 fast core fuel storage racks (4 fuel assemblies per rack)
- 10 control rod storage racks (6 control rods per rack)
- 1 channel stripping machine
- 1 emergency condenser
- 1 steam dryer
- 1 steam downcomer

Space is provided in one corner to place a shipping cask while it is being loaded.

After a storage period, the radioactivity and heat release rate from a batch of spent fuel have decreased to a point where transfer to the reprocessing facility is economically feasible. At this time, the shipping cask is lowered into the storage pool using the reactor building crane. Spent fuel assemblies are grappled into the cask and the cask lid is lowered into place. The cask is lifted out of the pool, the lid is fastened and the exterior surface of the cask is decontaminated. The cask is then placed on a waiting truck or railroad car.

Equipment for the channel stripping operation is mounted at the end of the storage pool away from the reactor. A spent fuel assembly with channel is placed in a fixture attached to the pool wall using the fuel grapple on the refueling platform. Bolts attaching the channel to the upper tie plate of the fuel assembly are removed under water with an extended wrench. The light weight channel is lifted off of the fuel assembly using a special hand grapple on a pole. The spent fuel assembly is transferred to a storage rack with the fuel grapple. The channel is inspected under water with the boroscope and, if reuseable, is lowered over a new fuel assembly, which has been placed in the channel stripping fixture. New attachment bolts and standoff springs are installed with the extended wrench and the fuel assembly is ready for loading into the core. The elevation of the fixture holding the fuel assembly or channel can be adjusted for these operations using a hand wheel connected to a chain drive built into the channel stripping assembly.

4.11.2 Maintenance

4.11.2.1 Control Rod Handling

Two types of control are used for the Mixed Spectrum Superheater: (1) the conventional poison rods in the thermal boiler, thermal buffer, and in the region between the fast and thermal buffers; and (2) the moving fuel elements in the superheating section. The control elements in the fast core and thermal buffer are driven from above by twin-screw drives. The remaining control rods (thermal boiler and between the two cores) are locking piston drives attached to the bottom head of the reactor pressure vessel. Control rod life depends on the time the control rods spend in the high flux region of the core. Control rod strength is normally checked during critical testing after each change in core loading. In the thermal core, when replacement is indicated, the fuel assemblies surrounding the control rod are removed. The control rod is attached to the drive mechanism by a spring-loaded coupling that can be actuated from above the core to release the control rod from the drive mechanism. The control rod is then grappled like a fuel element and moved to the storage pool. Similarly, the top-mounted control rods can also

be decoupled from their drives and are handled in a manner similar to the fuel.

4.11.2.2 Control Rod Drive Mechanism Removal and Replacement

Infrequent drive maintenance is required to inspect wear surfaces and to replace seals and other worn parts. For the bottom-mounted drives, drive removal and replacement is performed in the room below the reactor vessel during a plant shutdown. The control rod is withdrawn from the core until a seal surface on the coupling closes off the top of the control rod drive thimble. The coupling is actuated either from above the reactor, if the vessel head is removed, or from below the reactor, by using an activating cylinder and external jack screws attached to the drive or a similar scheme to accomplish this purpose. Water in the drive is drained, drive mounting bolts are removed, and the drive is lowered to a handling dolly with a hoist attached to the surrounding structure. The procedure is reversed to reinstall the drive.

In case a drive becomes jammed in an extended condition, it can be removed by using essentially the same procedure described above. The control rod is removed by uncoupling from above. The thimble is lowered over the extended control rod drive and sealed against the control rod drive thimble at the bottom of the reactor vessel. The drive is then drained and lowered in the conventional manner.

The uncoupling mechanism for the top-mounted control drives have not been worked out in detail as yet. Appropriate means will be developed for uncoupling the drive mechanism from the control element connecting rods prior to the removal of the vessel head.

4.11.2.3 In-Core Flux Monitor Servicing

The lifetime of the in-core flux monitors varies with the reactor flux level at a particular core location. Since the flux monitor assemblies are in guide tubes that penetrate the lower head of the reactor vessel and extend upward between fuel assemblies, they do not have to be removed during refueling and will be disturbed only when replacement is required. The equipment and procedure used for replacement of the flux monitor assemblies is shown on Figure 4.11.3.

4.12 Fuel Pool Cooling System

The Fuel Pool Cooling System is provided to remove the heat generated by spent fuel elements which have been removed from the reactor and are awaiting shipment to a reprocessing plant. It will also remove heat added to the pool water by the operation of the emergency

condenser coils. The system includes recirculating pumps, in-line filters, and a piping and valve system which interconnects the fuel pool, the reactor, and the shutdown heat exchangers. The latter equipment serves as the fuel pool heat exchangers.

Pool water temperature is maintained below 100°F by circulating water through one of the shutdown heat exchangers.

Each of the pool recirculating pumps has sufficient capacity to provide a complete change of the pool approximately once every 12 hours.

Normally, water is taken from the fuel storage pool, filtered and/or cooled and returned to the pool. Cooling is required only when fuel is in the pool. At other times only the filter is required to maintain water clarity. In order to maintain clarity of water in both the pool and the reactor during refueling, water is withdrawn from the bottom of the reactor, filtered, and returned to the fuel storage pool.

4.13 Vents, Drains, and Blowdown System

Pressure parts containing or conveying radioactive fluids will be properly vented to the condenser, blowdown tank, or stack. Various tanks and sumps will be provided to collect all radioactive drains. All safety valves will normally be vented to the blowdown tank as will the discharges from all seal relief valves.

RUPTURED FUEL SIPPER

FIGURE 4.9.1

REFUELING SYSTEM

67329

FIGURE 4.11.1

66474

GRAPPLE

FIGURE 4.11.2

PROCEDURE

- DISCONNECT LEADOUT CABLES & CALIBRATION TUBE EXTENSION.
- UNSCREW FLOATING UNIT & WITHDRAW CABLES FROM REACTOR THIMBLE UNTIL RADIOACTIVE PORTION APPEARS.
- CUT OFF EXPOSED CABLE & FASTEN END OF RADIOACTIVE CABLE TO DISPOSABLE REEL IN SHIELD POSITIONED BELOW THIMBLE.
- CLOSE SHIELD DOOR & ENERGIZE MOTOR DRIVE TO WIND UP RADIOACTIVE CABLES & CHAMBERS ON REEL.
- TRANSFER SHIELD CONTAINING FLUX MONITORS TO WASTE DISPOSAL AREA.
- DISCHARGE REEL & COVER CONTAINING FLUX MONITORS TO DISPOSAL PIT.
- INSTALL NEW DISPOSABLE REEL & COVER IN SHIELD FOR NEXT FLUX MONITOR REMOVAL OPERATION.

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FIGURE 4.11.3

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5.0 RADIATION CONTROL SYSTEMS

5.1 Shielding

5.1.1 Shielding Criteria

The basic criteria used to develop the shielding necessary for various systems and areas of the plant are a function of source strength and the degree of occupancy anticipated. In general, shielding is chosen to provide adequate protection as follows:

<u>Expected Occupancy</u>	<u>mrem/hr</u>
Substantial to continuous	1
Less than 10 hours per week	6
Less than 5 hours per week	12

5.1.2 Arrangement and Size

Layouts showing materials, locations and approximate dimensions of shielding are included on the reactor enclosure arrangements. The material used for biological shielding is reinforced concrete with an approximate density of 140 to 150 lbs. per cubic ft.

5.1.3 Shielding Functions and Accessibility

a. Reactor Containment

In general, the containment area of this plant is accessible, with suitable provision for controlling the access of personnel in or out. Operating floors, corridors, and certain rooms are accessible at all times during normal operation. These areas have been shielded to reduce the radiation level to less than 6 mrem/hr. Certain cells or areas listed below would not normally be entered during operation.

1. Area within biological shield of reactor vessel.
2. Sub-pile room.
3. Pipeways enclosing primary steam piping.
4. Reactor clean-up demineralizer system tank cell.

Refueling of the reactor is accomplished under the necessary amount of water for shielding. The typical radiation level in the operating area above the water shield is approximately 1 mrem/hr.

When the plant is operating and the refueling shield tank moved, the shielding floor between the reactor vessel head and the work

area for a cask removal provides shielding protection for operating personnel. During this condition, the typical radiation level in the work area is approximately 1 to 5 mr/hr.

b. Control Room

The control room will be located so that accessibility is normally unlimited. That is, the shielding, if any, is arranged to provide for radiation levels below the maximum permitted for unlimited occupancy. The shielding and accessibility must permit 8 hours occupancy in the event of a "maximum credible accident".

c. Turbine Building

The specific shielding requirements for the turbine building will depend on the arrangement of equipment. General Electric experience with boiling water reactors indicates, however, that the following thicknesses of ordinary concrete will be sufficient to allow continuous occupancy of the areas outside the shielded compartments:

<u>Equipment/Area</u>	<u>Shield Thickness (Feet of Concrete)</u>
Condenser	2-1/2 - 3
Hotwell	2-1/2 - 3
Extraction Lines	4
Steam Lines	4
Off-gas Lines	4
Air Ejector	4 - 4-1/2
Condenser Demineralizer	3
Feedwater Heaters	4-1/2
Turbine Floor	None required if adequate distance is available. 100 ft. will reduce the activity to 0.5 mrem/hr.

These shielding specifications include no allowance for additional radioactivity attributed to corrosion and fission products picked up by the steam in the superheater and transported to the turbine and condenser. These products, if they plate out, may require additional shielding in selected places. This additional shielding cannot be specified at this time since the necessary information for even approximate sizing is totally lacking.

5.1.4 Conditions Under Which Maintenance of Primary Loop is Feasible

<u>Components</u>	<u>Conditions</u>
Emergency condenser	Plant not in operation.
Reactor vessel-head removal	Plant not in operation - contact to head and studs.
Reactor vessel - maintenance other than head	Plant not in operation - major maintenance inside reactor vessel requires removal of fuel to adjacent storage area and drainage of system water. Minor maintenance outside reactor vessel with portable shielding.
Recirculating Pumps (not applicable for 75 MWe plant)	Plant not in operation - isolating valve to reactor vessel closed and system drained (except reactor vessel). Minor maintenance outside recirculating pumps - plant shut down.
Unloading heat exchanger	Plant can be in operation.
Reactor cleanup demineralizer	Minor maintenance during operation.

5.1.5 Shielding Fission Product Release

In the event that fission products might be released to the cells housing the primary water loop components and steam system up to and including the hot well, adequate shielding must be available to reduce the radiation to tolerable levels. The normal shielding requirements for these areas are established by the high energy gamma radiation from short-lived Nitrogen-16. These shields are highly efficient for the much lower gamma energies emanating from combinations of mixed corrosion products and fission products that might be released. For example, in order for the radiation level transmitted through a shield wall to be doubled, a mixed fission product concentration about 100 times greater than the normal Nitrogen-16 concentration is required. Such mixed fission product concentrations are far greater than concentrations normally maintained in the primary water and steam system. Thus, ample shielding is available to attenuate radiation from fission products anticipated in the primary water and steam system.

Equipment which is subject to potential fission product contamination has provisions for introducing decontaminating solutions; decontamination is hereby possible in the event that residual radiation levels must be reduced to permit direct access equipment maintenance.

5.1.6 Controlling Radiations for Shielding

N¹⁶: Primary water loop and in-steam loop including feedwater heaters, turbine, condenser-hotwell and air ejectors.

Corrosion Products: In demineralizers and upon shutdown, in areas.

Fission Products: Where settling out or plating out is anticipated.

The reactor shielding is adequate for all neutron gamma radiations generated within the reactor core or structural materials or coolants of the reactor.

5.1.7 Biological Shield Cooling

The concrete shield around the reactor vessel absorbs the major portion of radiation emanating from the vessel. To prevent this heat from causing over-heating and damage to an uncooled shield, a cooling system is installed in the shield, based on the following criteria:

- a. Demineralized water is used as a cooling medium.
- b. The average temperature difference throughout the shield will not exceed 50°F, and no part of the shield will exceed 200°F.
- c. The maximum local temperature gradient will not exceed 20°F/ft.
- d. 100 percent excess capacity will be provided with means of plugging defective cooling coils.
- e. Valves to control coolant flow will be operable during reactor operation.

The shield cooling system consists of carbon steel cooling coils imbedded in the concrete shield immediately under the inside face. Several coils are used with provisions for cutting out defective coils. The coolant flow is provided by the demineralized water cooling system.

5.2 Reactor Enclosure

The complete nuclear steam supply system, including the reactor, auxiliary components, and fuel handling and used fuel storage facilities are installed in a 72-foot-diameter cylindrical, vapor-tight containment vessel. (See Drawing No. 142F239.) Two access pressure locks are provided. One large primary lock is for access during construction and for general use during plant operation and maintenance. A second smaller lock will provide personnel access.

5.2.1 Design Data and Codes

The reactor enclosure and associated structure will be designed and constructed in accordance with specifications and requirements of the ASTM, ASME, ACI, AISC, NBFU, local building codes where applicable, and with the AEC Manual Chapter 6300. Complete borings and laboratory tests will be required to verify the locations, extent and quality of rock, and to verify the bearing values.

The ASME code for unfired pressure vessels governs the design of pressure parts. A 5% allowance for horizontal acceleration due to earthquake is included. A wind load of 15 pounds per square foot on the projected flat surface, and a snow load of 30 pounds per square foot, are allowed for.

5.2.2 Design and Test

The dimensions of the containment vessel are the minimum required to contain the equipment, pressure ratings are based on containing the energy and materials released in the event of a rupture of the primary system. The thickness of the shell plate is less than 1 inch so that field stress relieving is not required.

The containment vessel is designed to withstand the pressure resulting from the maximum credible accident. It will be pneumatically tested. A test to determine the leakage rate in percent of the design pressure will be conducted; the leakage rate of the enclosure is not expected to exceed 0.5% per day of the free volume at the appropriate design pressure. The permissible percentage leakage will be established after review of the specific site conditions. Steel plate for the containment building will be ASTM A-201, Grade B, firebox quality, made to A-300 requirements.

5.2.3 Isolation Valves

All pipes penetrating the containment are provided with remote-operated isolation valves. As a general rule, lines entering the containment are fitted with a check valve immediately inside; leaving lines are provided with a motor operator at the first stop valve inside.

5.2.4 Other Closures

Electrical penetrations are sealed and made pressure tight; ventilation ducts are fitted with motor-operated butterfly valves.

Two pressure locks are provided for access. The main lock is sized to permit passage of the used fuel shipping cask, and is used for general access. The smaller lock is sized for personnel access only.

5.2.5 Ventilation, Cooling, and Insulation

Ventilation is designed to provide a minimum of one complete change of air every 3 hours. About 8700 scfm of ventilation air is required. The supply duct is fitted with filters and tempering coils to clean and either cool or warm the entering air. Two full-capacity exhaust fans, located outside the containment, discharge air through an exhaust duct to the stack. Distribution duct work and dampers are provided.

The ventilation system is not designed for complete heat removal at full power, and unit ventilating sets with cooling coils are installed where required. The cooling system is designed to maintain a maximum of 130°F in non-habitable spaces, 110°F in spaces intermittently accessible, and 80°F where continuous access is required.

To reduce solar heat absorption, external insulation is required on the steel containment building. The insulation will also prevent the vessel metal temperature from falling below 32°F. This provides an adequate margin with respect to brittle fracture of the vessel.

5.2.6 Post-incident Cooling

To control and reduce the pressure within the containment caused by a rupture of the primary system, an independent emergency cooling system is provided. This system sprays water into the containment vessel and condenses the vapors released from the primary system.

Two full-capacity pumps are furnished. To prevent flooding, these pumps will take hot water from the bottom of the containment and pass it through the post-incident heat exchanger before returning it to the spray nozzles. Both pumps are operated from the emergency power system, and cooling water for the exchanger is supplied from either the service water or fire main systems.

5.3 Decontamination

5.3.1 Coolant Loop

The reactor water system contains activated corrosion products which are dispersed in solution and absorbed on system surfaces. These activated corrosion products vary in half-life from 2.6 hours to 5.2 years for the cases of Manganese-56 and Cobalt-60, respectively, and are the governing factors limiting contact maintenance after shutdown. Further contamination may be introduced in the reactor water system as fission product species from fuel cladding failures. Fortunately, in the case of the uranium dioxide fuel, uranium and the solid fission products are released to the reactor water at extremely low rates. Noble gas and halogen isotopes are released from the fuel. The noble gases will be released via the off-gas system. However,

the halogens will be retained within the coolant and then removed in the reactor cleanup demineralizer.

The turbine and condensate systems are not seriously contaminated with reactor water corrosion or fission products from boiler fuel due to the high separation efficiency of the steam-water separator and the fact that the principal fission products are gaseous and, therefore, go through rather than deposit in the system. However, steam corrosion-erosion products and fission products from possible defective superheater fuel pass directly to the turbine and condensate system. These products may plate out and give rise to serious contamination. The magnitude of this problem, common to all steam cooled reactors, is ill-defined due to the lack of data on corrosion-erosion of stainless steel and Inconel by steam, indefinite cross-section for the $\text{Ni}^{60}(\text{n},\text{p})\text{Co}^{60}$ reaction and plate out characteristics of particles in turbine and condensate systems. The feedwater system is completely protected from contamination due to full flow demineralization of the condensate.

Provisions are incorporated in the design of the nuclear power plant for isolation and decontamination of equipment. Valved connections or blind flanges are provided on reactor coolant equipment to facilitate installation of decontamination equipment at such a time as deemed necessary from maintenance considerations. Further, sleeves are provided through shielding walls for later pipe routings, and space is reserved within the reactor enclosure for later installation of decontaminating equipment such as tanks and pumps.

The type of decontaminant to be used will depend on the type and degree of contamination, and will be determined at the time of use.

5.3.2 Fuel Shipping Cask

Fuel shipping casks may be decontaminated in the enclosure vessel by washing. A floor sump is provided in a guttered area draining to the waste disposal system. No special ventilation is needed.

5.4 Radioactive Waste Disposal

The deaeration system for a nuclear plant requires special consideration due to the relatively large quantities of gases to be handled, the radioactivity of these gases, and the potentially explosive situation created by the presence of radiolytic hydrogen and oxygen.

5.4.1 Off-Gas System

(a) Objectives

The objectives of the off-gas system are:

1. Provide sufficient decay time for N^{16} and O^{19} produced in the reactor so that activity from these elements will not result in significant radiation exposure to plant personnel or the environs after release to the atmosphere via the main stack.
2. Provide for controlled release and dispersion of fission gases, primarily iodine, xenon and krypton, which may be released in significant quantities due to fuel element rupture.
3. Keep radioactive particle escape to the atmosphere below safe limits.
4. Minimize the safety hazard due to the presence of hydrogen and oxygen in explosive concentrations in the off-gases.

(b) Main Condenser-Air Ejector Off-Gas System

This system should provide a 30 minute hold-up of gases prior to release from the stack. A minimum decay time of five minutes is required to allow decay of N^{16} and O^{19} to negligible radiation levels. The remaining hold-up time provides for the operator to take suitable action to reduce or otherwise control the release of noble gases should this occur at high rates. (See Figure 5.4.1)

To control any radioiodine released to the off-gas system, charcoal filters are provided. These have an efficiency of 99.998%. These are preceded by a preheater to dry the gases to about 80% humidity to prevent wetting of the charcoal filters. A glass wool filter is provided to remove particulate matter. It has an efficiency of 99.99% and is treated to withstand moisture condensing from gas streams.

A potential hazard in the off-gas system exists after the air ejectors due to the presence of an explosive mixture of radiolytic hydrogen and oxygen. The system is designed to withstand an explosion should one occur. Also, means are provided to automatically (on high pressure or temperature) quench the explosion by addition and dilution with steam. Actually, the probability of an explosion occurring is very low since the off-gas system is closed and no source of ignition, spark or heat is present. The gas is saturated with water vapor so no static spark should result.

The greatest explosion hazard exists when the system is opened for maintenance or inspection. This can readily be handled by proper precautions and purging of the system prior to its being opened.

Control of fission gas release rates as well as of other gases is afforded by several continuous radiation monitors on the process off-gas system. These monitors provide a continual record of the radioactivity of gases released to the off-gas system. In the event that gas rates exceed a continuous safe discharge limit on the monitor, a valve in the off-gas can be manually closed to retain the gases. For a high level, this valve closes automatically. Valve closure requires a plant shutdown and correction of the cause of high release. After suitable decay, the retained gases are released and plant operation continued.

Continuous monitors on the stack provide a record of the total activity of all air and gases released from the stack. A particulate sampler is also provided on the stack to detect any significant release of radioactive particles. These monitors serve to indicate any changes or trends in release rates of radioactive gases or particles as well as total amount released. Means are also provided for periodically measuring the concentrations of individual radioisotopes so that they may be related to appropriate limits.

(c) Gland Seal Condenser Off-Gas System

This off-gas system provides 90 second hold-up between the gland seal condenser and the stack for decay of N^{16} to a negligible radiation level. Gases removed from the condenser by the vacuum pump at startup will be routed to the stack via the gland seal off-gas line.

5.4.2 Liquid Wastes

The liquid radioactive waste system collects and processes both normal and non-routine liquid radioactive wastes from the reactor power plant. Wastes are routed to various collection tanks according to expected purity, processed by filtration and demineralization or concentrated by evaporation, and either returned to condensate storage for reuse or disposed by storage in underground storage tanks. The waste disposal flow diagram (Drawing No. 141F949, Figure 5.4.2) shows all the waste routings, tanks, equipment, etc. Sizes, numbers and types of tanks and equipment are shown.

The Clean Waste Tank receives wastes from equipment and sample drains that are expected to be high in activity but low in dissolved solids. Refueling shield tank water is routed to this tank, as is reactor blowdown water via the clean-up system. Wastes from this tank are normally processed through the clean-up filter and demineralizer, but excessive wastes are processed through the waste filter and demineralizer.

The Floor Drain Collection Tank receives wastes from floor drains and demineralizer backwash that are expected to be relatively low in activity but high in dissolved solids. These wastes are processed by the waste filter and demineralizer, then collected in a Waste Holdup Tank, sampled, and pumped to condensate storage. The waste demineralizer resin can be regenerated if the activity is not too high; alternately the spent resin may be discarded to storage.

The Waste Neutralizer Tank receives demineralizer resin regeneration wastes from the condensate demineralizer system. Wastes from other collection tanks that are abnormally high in activity are routed to this tank. These wastes are concentrated in the waste concentrator, then stored in the Concentrated Waste Storage Tank.

During normal operation, the water from the fuel pool is circulated through a filter to remove particulate impurities. During refueling, the water in the reactor and the refueling shield tank are also filtered.

All filters are located in the Rad Waste Building to facilitate filter sludge handling. The slurry is discharged from the filter to the Filter Sludge Tank, fed to the waste centrifuge, where excess water is removed, and returned to the Clean Waste Tank. The solid sludge from the centrifuge is discarded to the concentrated waste storage tank for storage.

The proposed system permits considerable flexibility in operation. Tank and treatment capacity is sized to handle peak loads which may be considerably above the expected daily averages. There is sufficient cross-connecting to provide for emergency surges as well as present a choice of process steps in the event of abnormal activities at leaks.

5.4.2.1 Expected Waste Quantities and Activities

Table 5.4.1 lists the main contributors of radioactive wastes with the expected flows and activities and process capacities. The values listed in this table are estimates based upon BWR experience and are, at best, order-of-magnitude estimates. Only a cursory examination has been made to determine the effect on waste quantities of the superheating core. It is believed that the quantities shown are indicative of the magnitudes to be expected, but the sole purpose of including them here is to provide a basis for making the cost estimate for the plant. The detailed design studies will proceed from this point.

In Table 5.4.1, where a range of values is given for the flow, the lower is the expected daily average flow while the higher

value represents the maximum expected on a given day. The lower value for the estimated activity represents only corrosion product activity and the higher value represents a large number of fuel leaks present. The last column lists the process capacity of the systems, allowing time for sampling, backwashing and other operations. Processed wastes from the Clean Waste Tank along with the clean-up flow will normally be returned to the condenser hotwell. Processed wastes from the Floor Drain Collection Tank will be returned to condensate storage, provided the conductivity is less than 1 micromho and the activity is less than 10^{-3} $\mu\text{c}/\text{cc}$. Wastes from the Waste Neutralizer Tank will be concentrated and stored in underground tanks designed with a six month capacity.

Although wastes are not normally discharged off plant, provision is made for a route from the waste system to the condenser water discharge canal. This is necessary for the disposal of non-radioactive water from the reactor flushing operation before initial startup. It may also be used in the event that makeup exceeds normal plant water losses.

Following a refueling operations, the water from the refueling shield tank is pumped via the fuel pool recirculation pumps to the waste system. Here the flow is immediately processed through both the waste demineralizer and the cleanup demineralizer and then routed to condensate storage.

5.4.2.2 Construction

To prevent leakage to ground, the subsequent contamination of ground water, secondary containment is provided for radioactive waste piping and equipment. Means of detecting overflows and leaks are provided. Containment is provided by locating pipes and equipment inside buildings, dikes or concrete vaults. Exceptions include lines carrying treated wastes having low enough activity to preclude significant ground contamination.

Tanks are of welded construction and materials are suitable for the intended service. Carbon steel is generally used. Stainless steel is used where corrosive conditions require its use. Piping is primarily welded, although some flanged connections are used. No threaded fittings are employed in waste piping.

5.4.2.3 Operation

The liquid waste system is controlled from a local panel in the waste building. Operation is semi-automatic remote rather than at the equipment, because of radiation and the remote location of most equipment behind shielding.

5.4.3 Solid Wastes

Solid radioactive wastes result from operation and maintenance of a nuclear power plant. Means for safe handling and disposal of these wastes are necessary to assure proper control and to prevent spread of contamination.

(a) Sources and Types of Wastes

The following are typical of potentially radioactive solid wastes:

1. Spent radioactive resins from demineralizers.
2. Air filters from off-gas and ventilation systems.
3. Miscellaneous paper, rags, etc., from contaminated areas.
4. Contaminated clothing, tools, and small pieces of equipment which cannot be economically decontaminated.
5. Solid laboratory wastes such as contaminated glassware, etc.
6. Used reactor equipment such as spent control rods, fuel channels and in-core ion chambers.

In addition to the above, which are generated more or less regularly, there may be occasionally a relatively large piece of contaminated equipment which could be more economically replaced than decontaminated and repaired.

(b) Methods of Handling and Disposal

The general plan is to temporarily hold the solid waste on the plant site in a separately controlled area until suitable amounts are on hand for shipment to a permanent off-site disposal site. However, used reactor equipment may be first stored for several years in the fuel storage pool to obtain optimum decay before removal to final storage. Shielded containers will be required for these items.

Most of the solid wastes are not so radioactive as to preclude handling by contact. They will be collected in a standard size container, so final storage space is used most effectively. The containers are located in appropriate zones around the plant as dictated by the volumes of wastes generated during operation and maintenance. The containers (fiber drums, cartons or boxes) will be periodically monitored during filling, so that the contents do not exceed a prescribed maximum before disposal (a reading of 50-100 mr/hr would be a maximum).

Where possible, wastes which are compressible will be compressed in a conventional press and baling machine to reduce storage volume. Filtered ventilation air flowing past the press will prevent spread of particulate contamination which may result from compression of solid wastes. Compressed or not, solid wastes will be stored in a restricted area until off-site disposal is accomplished.

Equipment too large for the above handling and which may require disposal on occasion will be handled as a special case at the time of disposal. Since the frequency and need for disposal of large equipment are unknown and will probably be quite infrequent, the provision of disposal facilities in advance is not justified. Disposal of such equipment will depend upon the radiation level, transportation facilities and available burial or disposal sites.

TABLE 5.4.1Expected Flow

<u>Component</u>	<u>Average Daily Gal/Day</u>	<u>Maximum in Any Day Gal.</u>	<u>Estimated Activity $\mu\text{c/cc}$</u>	<u>Maximum Process Gal/Day</u>
Clean Waste Tank	8,000	12,000	5×10^{-3} to 1	53,000
Floor Drain Collector Tank	4,000	12,000	10^{-4}	60,000
Reactor Encl. Floor Drains	500	3,000	10^{-4}	
Turbine Bldg. Floor Drains	500	3,000	10^{-4}	
Cond. Demin. Backwash	3,000	10,000	10^{-4}	
Waste Neutralizer	1,000	4,000	10^{-4} to 10	5,700
Refueling Shield Tank		65,000	10^{-3}

FIGURE 5.4.1

SCHEMATIC DIAGRAM OFF GAS DISPOSAL SYSTEM

65173

NOTES:
 1. SPENT RESIN FROM WASTE DEMIN. IS SENT TO COND. DEMIN. SYSTEM FOR REGENERATION OR TO SPENT RESIN STORAGE TANK.
 2. ALL TANKS & EQUIP. TO BE PROVIDED WITH SUITABLE VENTS, DRAINS, & OVERFLOWS.

FIGURE 5.4.2

6.0 CONTROL AND INSTRUMENTATION

6.1 General Plant Control

The control and instrumentation system enables the operator to start up, operate and shut down the reactor plant, and to protect plant personnel and equipment. The system consists of the necessary control and instrumentation equipment incorporated in centralized control consoles and panels. Conventional power plant practice is generally followed, except where the operating characteristics and radiation protection requirements of the nuclear plant require special consideration.

The main control room contains all the control and instrumentation essential to the operator during normal and abnormal plant operation. Lesser instrumentation, for which there is no necessity for constant or immediate visual observation, is located in the auxiliary control room, near the main control room, or on local control panels in the equipment area.

Instrumentation and controls for the turbine-generator, feedwater heaters, and other portions of the power plant are similar to conventional fossil-fueled power plants. These include normal safety features such as turbine overspeed trip, protective relaying and control, and indication and recording of critical temperatures, pressures, and flows.

With the exception of the In-Core Monitor System, the nuclear instrumentation portion of the plant has been designed for eventual operation at the 150 MWe power level. For 150 MWe operation, additional in-core chamber assemblies and associated instrumentation and wiring would be required for effective control.

6.2 Plant Operation and Control

For base load operation, this plant will have the following main control features:

1. System steam pressure will be regulated automatically by controlling the turbine control valves.
2. Power (hence, steam flow) in the boiling region of the reactor core will be controlled manually by positioning control rods (poison).
3. Superheat steam temperature will be controlled manually by controlling the power in the superheat region by positioning control rods (fuel).
4. Load rejections will be absorbed by a full-capacity steam bypass valve to the condenser.

6.3 Pressure Regulator and Bypass System

The turbine control mechanism includes a conventional governor and an initial pressure regulator. The pressure regulator controls the turbine admission valves to maintain automatically the system pressure. This regulator mechanism, shown sketched in Figure 6.3.1, is similar to that used for the Dresden plant which can maintain the primary steam pressure of 1000 psi within 2 psi. The turbine speed governor output signal is set some amount above the pressure regulator so that it does not normally control the turbine admission valves. The steam bypass valves are normally closed, and all reactor steam flow is through the turbine.

With the control system set up in this manner, the turbine is a slave to the reactor since it will follow the reactor output rather than system load changes. Load changes are met by the operators by positioning reactor control rods to change reactor power. The new steaming rate is accepted by the turbine through action of the pressure regulator on the turbine control valves.

Should the maximum allowable control valve opening be reached (load or power limit), the pressure regulator will open the bypass valves to divert the excess steam flow to the condenser. One way this situation might occur would be for the operators to bring up the reactor power beyond the full load power rating of the turbine. (The condenser is sized at 125% of turbine capacity to accommodate such conditions.)

A load rejection which causes turbine overspeed (beyond 102%) will cause the speed governor to override the pressure regulator, closing the control valves and opening the bypass valves. The full capacity bypass valves operate as fast as the control valves; therefore, a full-load turbine or generator trip-out can be accommodated with minimum disturbance to system pressure. Partial rejection may also occur; for example, if the plant loses its transmission tie-line leaving only a local load, or the interconnected system frequency goes up due to a sudden decrease in system load. In this case, the speed governor will partially close the control valves while opening, also partially, the bypass valves. The pressure regulator then will control the bypass valves to maintain set pressure.

An emergency (overspeed) governor will limit the maximum turbine speed to 110% of rated by closing both the stop valves and the control valves. As back-up, an emergency trip on the speed governor will also close both stop and control valves at 112% of rated speed. In either case, closure of the control valves causes automatic opening of the bypass valves. This can be seen in the simplified mechanical schematic diagram of Figure 6.3.2.

This diagram shows the interrelation of the various parts of the regulating system. The spring on the beam linkage normally holds the bypass valves closed. The pressure regulator then opens and closes the control valves in response to a pressure signal. Should the beam rise to the adjustable stop (preset for a maximum control valve opening) then the bypass valves open and close in response to the pressure regulator. The speed governor can move the linkage when turbine speed exceeds 102% of rated. The action of the governor is to push down on the beam, causing the control valves to close while opening the bypass valves. The pressure regulator then controls the bypass valves.

6.4 Prevention of Accidental Control Rod Withdrawal

To assure that the protection system controls are set properly and that certain control rod withdrawal patterns are avoided, interlocks are provided to prevent control rod withdrawal unless certain conditions are met. The following conditions must be satisfied before control rods can be withdrawn for reactor startup:

1. All three power range flux amplifiers (picoammeters) recording BWR flux levels must be set at their lowest operable range or two of three picoammeters must be reading greater than 5% on a 0 to 150% scale if set above the lowest range. (The picoammeters signal a reactor scram at a scale reading of 120%). The range switches on the picoammeter also activate the startup and intermediate instrumentation period protection channels when set down to startup levels.
2. All scram accumulator tanks for both locking piston and twin-screw drives must be up to operating pressure.
3. The coupling controls next to the superheater core and the controls in the superheater section cannot be moved unless a) the superheater section is unflooded and b) minimum coolant flow is established.

Interlock controls will also be provided on the locking piston drives to prevent withdrawal during refueling. The twin-screw drives are attached to the top head and will be unlatched with their control elements in their least reactive position prior to removal of the head.

6.5 Control Rod Hydraulic System Instrumentation

The operating conditions for both sets of control rod hydraulic systems are continuously monitored by pressure, level, and position instruments. These devices transmit signals to the control room to indicate system pressures and to annunciate inlet and outlet shutdown valve positions, loss of accumulator air pressures, water level in the air side of the accumulator, and dump tank water levels. These devices provide continuous indication or alarm of off-normal conditions of the system.

6.6 Liquid Poison System Instrumentation

6.6.1 Master Injection Controls

Injection of the liquid poison solution held in the storage tank is a manual operation which must be initiated by the reactor operator. Two master control switches are provided to open the two in-line solenoid valves. Two pump controls are provided for pump operation. The main pump has an a-c motor drive and the backup emergency pump has a d-c battery-powered motor drive. Valve position is indicated by pilot lights. Pump operation is indicated by pressure on the indicator at the pump located in the turbine building.

6.6.2 Injection System Test Control

The pumps may be operated with control valves closed through a relief valve by pumping back to the liquid poison storage tank. Valve leakage tests may be performed by manual valve operation and pump operation without injecting poison into the reactor. The two valves in series prevent accidental discharge of poison into the reactor.

6.6.3 Process Instrumentation

The liquid poison system is continuously monitored by process instrumentation to indicate operating conditions, standby conditions and accidental leakage of system valves. The electric immersion heaters which heat the poison solution are automatically controlled to maintain the set temperature. Two pressure control valves maintain a minimum of 1350 psig in the tank. High and low temperatures, low pressure, and low liquid level are annunciated in the control room.

6.7 Reactor Clean-up Control System

The reactor clean-up system is placed in operation manually and normally remains in service at all times except for periods of maintenance or resin sluicing. The temperature of the reactor clean-up water leaving the non-regenerative heat exchanger is recorded on the main control board, with abnormally high temperature annunciated. The high temperature switch also shuts off the clean-up pump to protect the clean-up demineralizer from high-temperature water.

The clean-up flow rate through the demineralizer is recorded and integrated on the turbine-generator board. The rate is controlled by remote manual regulation of the clean-up pump discharge flow control valve. An annunciator provides early warning of the impending completion of a pre-set flow cycle, at which time the resin will require replacement.

The differential head across the demineralizer is indicated with high-head annunciation. Conductivity of the demineralizer effluent is recorded on the reactor board. High conductivity is annunciated to indicate a major leakage of contaminants.

6.8 Reactor Vessel Instrumentation

Instrumentation applied to the reactor vessel monitors the vessel wall temperature, the internal steam pressure, the coolant liquid level, and the integrity of the vessel head "O" ring seal. The seals at the superheater outlet are also monitored for leakage to indicate the condition of the seals.

Reactor pressure and water level are measured by two separate and independent systems. The signals from the instrumentation are used for the safety system, reactor water level control, and pressure indication, and are transmitted to the control room for indication, recording and annunciation of deviations from normal.

6.8.1 Temperature

Metal temperatures of the pressure vessel wall are measured by thermocouples at several positions, axially and circumferentially, around the vessel, including nozzles and the head flange. The information is recorded in the main control room and is used to determine and monitor the allowable rate of reactor heating and cooling.

6.8.2 Pressure

Two separate and independent sets of reactor vessel fluid taps will be provided for reactor fluid measurements. Pressure signals from the taps will be independently transmitted to the control room for recording and indication. Two pressure switches on each of the fluid taps are connected into the reactor safety circuit to provide a scram signal upon excessive pressure. An overpressure alarm function, provided by alarm contacts on the recorder, annunciates an over-pressure condition below the pressure setting of the overpressure scram switches.

6.8.3 Liquid Level

The liquid level in the reactor vessel is monitored through both of the reactor vessel fluid taps mentioned above. Liquid level signals from each of the taps are transmitted to the control room for recording and indication. Dual liquid level switches, on each of the taps, are connected into the reactor safety circuit to scram the reactor before the water level gets so low that there

would be danger of uncovering the top of the chimney and stopping reactor coolant recirculation. Additional liquid level switches indicate an auxiliary core coolant system in the event of exceptionally low water level. High water level, which might flood the steam drier or drastically increase water carryover activates liquid level switches which scram the reactor. Thus, the possibility of damaging the superheater fuel due to excessive water carryover is minimized.

In addition to the liquid level indicator in the boiler, provisions are made to indicate water level in the fast core region and in the hot well of the fast core drain system. The signals from these transducers will go to interlocks as well as to indicators to prevent starting up the fast core while it is flooded. Interlocks on flow measurements will also be provided to assure that sufficient steam flow is established before the fast core is started up.

6.8.4 Leak Detection

Continuous monitoring of the seal integrity at the flange connection of the reactor vessel head is provided by connecting a drain line to the flange face between the flange "O" rings. The drain line is normally closed with maximum pressure established by a low pressure (approximately 30-50 psig) relief valve. Leakage from the reactor vessel increases the pressure in the line and trips the pressure switch which is set below the relief valve setting. A bypass needle valve around the relief valve and a local pressure gage permits an operator to test the line to determine the magnitude of the leak. Rapid drain line pressure decay upon opening of the bypass needle valve indicates an insignificant leak.

The seals on the superheater outlet line will be similarly instrumented to provide continuous monitoring of the seals.

6.8.5 Differential Pressure Instruments

Four differential pressure transmitters are provided. One senses the pressure drop across the boiling core, while a second senses the pressure drop across the superheater. A third is connected across the upper section of the reactor vessel and is used as a water level sensor during filling operations. These instruments transmit to common three-pen recorders in the main control room.

A fourth differential pressure transmitter is connected across the fast core to be used as a water level sensor when the fast core is flooded.

6.9 Plant Control and Instrumentation System

6.9.1 Reactor Instrumentation and Control

There are two separate reactor pressure transmitters. One transmits to an indicator on the console in the main control room and the second to a recorder located on a panel in the main control room.

The water level in the reactor is controlled by a three-element level control system. This control system uses the measurement of steam flow, feedwater flow, and reactor water level. Signals proportional to each of these measurements feed into the control systems. Normally, the steam flow signal equals that of the feedwater. Any variance in these results in a correcting signal to the feedwater control valve. The water level measurement adds or subtracts from this signal to the control valve if the detected level varies from the designed range. A bias signal, proportional to steam flow, is applied to the level measurement signal so that the water level control set point varies with load. This gives programmed level control. For high loads, the level stabilizes at the upper limit to provide volume for shrinkage on any load decreases. Conversely, at low loads, the level is maintained at a lower limit to allow for water volume swell on load increase.

A manual-to-automatic selector station, located in the center of the main control console, gives the operator means of switching from automatic to remote manual control of feedwater.

The water level, steam flow, and water flow rates are recorded in the main control room. Feedwater flow is also integrated.

Two level transmitters are supplied to transmit reactor water level to the control room. One of these is a duplicate of the unit used with the three-element control system and serves as an operating spare. By operating a switch, the operator may select which transmitter is connected to the feedwater control system. A third level transmitter serves as an additional backup reading for water level. There are also four low level and four high level sensing switches which cause reactor shutdown should the water level fall below or rise above the safe operating point. The high level switches are provided to prevent accidental flooding of the fast core during operation.

6.9.2 Superheater Temperature Indication

As a check on the flow distribution in the superheating core, the temperature of the steam at the exit of each fast core fuel bundle will be recorded. Accordingly, thermocouples are provided

in each fuel assembly to give a continuous indication of steam temperatures across the exit of the fast core. The leads from the thermocouples are taken up through the spacers placed around the fuel channels to the top of the fast core. From there the leads are passed up through a tube in the steam downcomer and out through the vessel wall.

The signal from these thermocouples will be used to scram the reactor on superheat overtemperature, if a preset number coincidentally indicate high temperature. The steam temperature distribution will also provide a means of determining the fast core radial flux distribution. The temperature of the superheated steam is also sampled downstream outside of the pressure vessel where three thermocouples are arranged for convenient replacement. These are the primary process control indicators and temperature scrams from these sensors are set lower than for the in-core thermocouples.

6.9.3 Reactor Shutdown Cooling System

Temperature elements feeding into a multiple point recorder are provided to measure the temperatures of reactor water influent and effluent from the shutdown heat exchangers and also cooling water above the boiler and superheater. The temperature of the cooling water effluent from each heat exchanger is similarly recorded.

Flow control valves are provided on the cooling water return lines to permit the proper distribution of cooling water between the boiler and superheater. The temperature of the water leaving the superheater fuel will be indicated by the thermocouples in the fuel assembly outlets to assure that proper cooling is obtained.

6.9.4 Emergency Condenser Instruments and Controls

At any time after the emergency cooling system has been placed into operation, and the reactor pressure brought below the pressure point at which the emergency cooling system was energized, remote, manual on-off control of the condensate valves may be used to regulate the rate of reactor cooling.

6.9.5 Emergency Dump Valve Control

In an emergency dump of reactor steam to the emergency condensation pool, the dump valves are programmed to open and then reclose in 10 to 30 seconds as desired. A positive and highly reliable actuator and control system must be developed for these valves to insure continuous cooling of the superheater. As a possibility, the valves might be hydraulically operated with a separate emergency accumulator to insure operation.

6.9.6 Reactor Enclosure Instruments

The instrumentation specifically for the containment enclosure consists of pressure-sensing devices. There are two transmitters, one with a low range of $-2\frac{1}{2}$ to $2\frac{1}{2}$ psig, and the second with a range of -5 to 45 psig. Both transmit to receiver indicators located in the main control room.

Four pressure switches are located externally to the enclosure and are designed to sense the pressure within the enclosure. These pressure switches are connected into the safety circuit for shutting down the reactor if the enclosure pressure should exceed ambient by an amount greater than that due to normal atmospheric pressure changes.

6.10 Shield Cooler

A temperature indicator, with contact for alarm, is provided on the shield cooler effluent water. A pressure gage ahead of a manually adjusted flow regulating valve is provided to allow calculation of the pressure drop across the cooler.

6.11 Reactor Neutron Monitor System, Drawing 612D629 (Figure 6.11.1)

An instrumentation system is provided to monitor the neutron level of the reactor from startup through full power. The instrumentation covers a range of nine decades in the three phases: startup range, period or intermediate range, and flux level or power range. Two different systems are used to independently monitor the BWR and the SH regions of the core. The system for the boiling region is described first.

6.11.1 Startup Range (Boiling Section)

With the initial fuel loading, a neutron source is inserted in the reactor core to assure a count rate of several neutrons per second. The startup instrumentation covers the range upward to about 1% rated power. Two channels of instrumentation monitor this phase of the operation. The primary neutron detectors are proportional counters housed in guide tubes adjacent and external to the reactor vessel wall about two feet above the mid-plane of the core. Neutron leakage through the vessel wall interacts with the boron-lined chamber of the proportional counter. The resulting ionization of the contained gas gives a count level.

By utilizing an advanced design Current Pulse Amplifier, (CPA), the pulses may be passed from the chamber directly to the CPA in the control room, thus eliminating any pre-amplifier in the vicinity of the reactor. The average rate of the series of pulses is measured on a count rate meter with a logarithmic scale in order to encompass six decades of measurements. The count rate is recorded continuously.

The gas in the proportional counter is also ionized by incident gamma rays from the reactor. These gamma pulses, which are of lower magnitude than those due to the neutrons, are eliminated from the counting circuits by pulse height selection in the log-count-rate meter. An additional circuit differentiates the log-count-rate, and indicates the reactor period at this low level on a period meter. The approach of a short period at this low level is annunciated.

6.11.2 Period Range (boiling section)

This phase of the instrumentation system, concerned with the rate at which the neutron flux is increasing, monitors and controls the reactor as it rises toward full power. Three channels of instrumentation cover this range. The primary detectors are gamma compensated ion chambers. These detectors have a gas volume sensitive to both gamma and neutrons and de-

liver a current in opposition to that from a compensating volume sensitive to gamma only, thus, eliminating the current contribution due to gamma. The detectors are housed in guide tubes near the reactor, similar to the startup channels. As the neutron count increases to about 105 cpm, the time average of this intensity of radiation results in a d-c current proportional to the neutron flux. D-c techniques are required to measure these currents of about 10^{-10} amps. The output of the gamma compensated ion chamber is passed to a Log N amplifier which indicates the logarithm of the input on a scale covering from 10^{-7} to full power. The output of the Log N amplifier is continuously recorded. A time derivative of the output is also indicated on a meter as the reactor period. An early warning will be indicated visually and audibly if either channel approaches a short reactor period. Reactor scram is initiated if the reactor does get on a short period.

6.11.3 Flux Level or Power Range(boiling section)

As the reactor reaches about 10^{-3} of full power, boiling begins and the steam voids cause the period measurements to fluctuate. In order to avoid spurious scrams, the period channels are disabled from the safety circuits at this point. However, two decades below this point, the power level instrumentation is functioning, and monitors the reactor through to full power. There are six channels of instrumentation to cover this range. The primary detectors are gamma compensated ion chambers, positioned in guide tubes at the mid-plane of the reactor and located externally to the reactor vessel. The output is fed into a power range picoammeter of fast response and stable characteristics. The power level is read on a meter indicating the percent of full power, and is continuously recorded. An early warning will be annunciated if any of the six flux level indicators exceed a preset percentage of over-flux. A signal to insert rods and shut the reactor down is indicated whenever one odd-numbered and one even-numbered power level indicator shows an excess of neutron flux.

6.11.4 Superheat Monitor System

The superheat region of the core will be monitored over the top five decades of operation, in two modes of coverage. This system is located in the reactor vessel. Two channels of period measurement will be provided covering this range. The detectors will be U^{238} lined which will respond to the fast spectrum present in the superheat region. They will be located in the region just above the center of the core. Mineral insulated cables will be used to carry the polarizing voltage

and signal through the cable penetration. A special irradiated polyethylene cable will be used beyond the penetration but still in the region of the reactor's external environment. The readout will be displayed in the form of Log N fast flux and period on the control console.

The upper $2\frac{1}{2}$ decades from .005 of fuel power to fuel power will be covered by three channels of linear power monitors. These will consist of miniature gamma chambers similar to those in current use in several BWR reactors as in-core fission chambers. The cooling is similar to the Log N fast detector system. The readout will be via flux amplifiers in the control room.

All five channels will give early alarm and also scram the reactor in a preselected logical one out of two or two out of three.

6.12 In-Core Flux Monitor System(boiling section) Drawing 985C743 (Figure 6.12.1)

The In-Core Flux Monitor System is used to determine the neutron flux distribution throughout the reactor core, in both the axial and radial plane, under full power conditions.

6.12.1 Flux Monitors (Boiling Section)

Flux is monitored at three elevations in eight locations in the boiler. The signal is carried in mineral-insulated, stainless-steel sheathed, cable. The cable is welded to the ion chambers to make a pressure-tight assembly.

The ion chamber assembly penetrates the top of the reactor vessel through a high-pressure connection. The chamber leads terminate in an electrical connector located at the end of the assembly outside of, and in close proximity to, the reactor vessel.

The cables from the electrical connectors on each assembly are collected at a distribution box, and routed to amplifiers and indicating instruments on the control panel. An integrated rod position and in-core flux level display is provided for the operator.

6.12.2 Wire Irradiation

A calibration tube runs concentrically with the in-core assembly and permits wire, of any desired metal, to be inserted in the 40 positions in the core. The wire is exposed during full power, withdrawn and counted. This wire irradiation provides additional flux profile data over the entire length of the active fuel in the

40 key positions. The equipment normally available in the counting room can be used to count the exposed wires for calibration. The axial power distribution in the superheat core is obtained from wire scans in locations in the thermal buffer immediately adjacent to the fast core.

6.12.3 In-Core Monitor System - (Superheat section)

The radial power distribution in the superheat core will be monitored by measuring the steam temperature at the outlet of the fuel. A resistance thermometer (RTD) will be housed at the exit point of each fuel bundle. Mineral insulated cable will take the resistance bridge signal out to the control room. Measurement of $1000^{\circ}\text{F} \pm 20^{\circ}\text{F}$ are known technology and in use, so that additional development work is not required.

6.13 Reactor Protection System

The reactor protection system is designed to protect equipment, plant, and personnel by rapidly inserting the control rods in the event of an accident or dangerous misoperation. Control signals, which initiate a rapid shutdown, originate from a variety of control and detection devices. These signals activate control circuitry to cause the reactor to be immediately shutdown, initiate operation of penetration closures, emergency cooling, and whatever other devices are necessary for safe plant shutdown.

The system consists of two independent "fail-safe" safety channels, both of which must be de-energized to produce a reactor shutdown or other safety system function. It is designed for long-life and maximum reliability, using proven, commercially available devices in coincidence circuits, so that, in most cases, the failure of a single component or power supply does not prevent a desired shutdown or cause an unwanted shutdown.

Each sensing element is continuously monitored so that an operation or failure (either continuous or intermittent) is clearly indicated and identified for quick and easy maintenance.

Wherever practical, the two channels are physically separated and clearly identified to minimize the possibility of maintenance personnel causing an accidental shutdown. The Reactor Protection System is a dual bus, static control system consisting of logic elements, power supplies, and power switches. The control elements are solid-state devices using silicon diodes and silicon transistors to perform logic and switching functions.

The plant conditions which are monitored by the reactor safety system and used to cause a rapid plant shutdown are summarized in Table 6.13.1.

TABLE 6.13.1
SAFETY SYSTEM FUNCTIONS

Conditions for Scram (Ventilation Ducts Closed and Turbine Tripped on Each Scram)	Open Dump Valves to Emergency Pool	Open Valves to Emergency Condenser	Close Enclosure Penetrations*
Short Reactor Period			
High Neutron Flux			
High Reactor Pressure	X	X	
High Enclosure Pressure			X
Low Water Level in Reactor			X
High Water Level in Reactor			
High Water Level in Rod Scram Dump Tank			
Low Condenser Vacuum			
Closure of Both Turbine Stop & Bypass Valves	X	X	
Closure of Steam Isolation Valves	X	X	
Loss of Auxiliary A-C Power	X	X	X
High Superheat Temperature			
High Ratio - Flux to Steam Flow			

* Valves in the steam lines to the turbine and condenser are not closed in this operation.

Typical block diagrams of the logic and switching system are shown on Drawing 612D731 (Figure 6.13.1). Each channel of the system is powered by two, 24-volt d-c power supplies. The logic elements are composed of diodes arranged to perform a desired logic and to provide a dual series output. The diode logic switching elements are operated in pairs with a common input and a two-transistor series parallel output. The power switches are operated in parallel, and are reset independently to reduce load in-rush current. During normal operation, the input to the power switches is held at a nominal 12 volts by the logic elements. To initiate reactor shut-down, the input to the power switches is dropped from 12 volts to a fraction of a volt, causing the silicon-controlled rectifiers to cut off. Actually, the trip level of the power switches is established at about eight volts.

The input to the logic element is a nominal 24 volts, d-c supplied through a pressure or level switch or supplied directly from one of the neutron monitoring instruments. If all inputs are 24 volts, all logic element output transistors are held in their conducting state, and the input to the power switches is maintained at 12 volts d-c. If any single input signal goes to zero, both transistors, in a given logic element, cut off and initiate a trip signal in the respective channel. If both channels of the reactor protection system are tripped in coincidence, the reactor is rapidly shut down.

The dual channel Reactor Protection System, utilizing series-parallel switching elements, provides a high integrity reactor control system with a very low incidence of false operation due to component failure. Any transistor or diode can fail in either a short circuit or an open circuit, and it will not cause a false scram or cause the reactor to fail to shut down if required. Furthermore, any logic element or power switch may be removed from the system without impairing the integrity of the Reactor Protection System.

A short explanation of each scram function listed in Table 6.13.1 is given below.

6.13.1 Scram from High Neutron Flux

This limits the instantaneous core power to a level well below that where any damage to the fuel could result.

6.13.2 Scram from High Reactor Pressure

This limits the rise in core power due to the positive pressure coefficient to a safe value. Since the pressure rise implies an interruption in steam flow, emergency actions (open valves to emergency pool and emergency condenser) are initiated.

6.13.3 High Enclosure Pressure

The rise in pressure indicates a major rupture of the high pressure system within the enclosure. Such a rupture could cause the water to drain quickly in the reactor vessel to the point of uncovering the core; the result would be fuel damage and release of radioactive material. Scramming the reactor minimizes the possibility of damaging the core and also the possibility of bursting the enclosure. Since there is the possibility of release of radioactive material, isolation of the enclosure is also initiated by closing critical enclosure penetrations.

6.13.4 Scram from Low Water Level in Reactor Vessel

As in the preceding scram condition of high enclosure pressure, a system rupture is the most likely cause for the water level to drop dangerously low. Therefore, the same reasons and actions apply.

6.13.5 Scram from High Water Level in Reactor Vessel

This prevents water from being carried by the steam into the superheat core causing subsequent erosion and thermal stresses. Flooding, of course, is the extreme condition this scram protects against.

6.13.6 Scram from High Water Level in Scram Dump Tank

This insures that the reactor is not operated with insufficient free volume in the control rod scram dump tank which would prevent rapid insertion of the rods.

6.13.7 Scram from Low Condenser Vacuum

This insures that the reactor is not operated without the main heat sink. Continued operation would cause escape of radioactive steam to the turbine room through ruptured relief diaphragms. However, the turbine stop and bypass valves do not close unless a drastic loss of vacuum occurs.

6.13.8 Scram from Closure of Both Turbine Stop and Bypass Valves

This minimizes the pressure rise due to the fail-safe action of these fast-acting valves. Since steam flow is interrupted, emergency cooling is also initiated.

6.13.9 Scram from Loss of Auxiliary A-C Power

This anticipates the scram that would result from a number of unsafe conditions caused by loss of pumps, instrumentation, etc. It is initiated by the fail-safe circuitry of the safety system which gets its power from motor-generator sets driven off the auxiliary A-C bus. Large inertias built into the M-G sets allow a few seconds auxiliary power interruption without scram. The fail-safe feature also initiates enclosure isolation and emergency cooling.

6.13.10 Scram From Closure of Steam Isolation Valves

Scram is initiated when the valves are only partially closed which will minimize the resulting pressure rise in the reactor. Emergency cooling is also initiated since steam flow is interrupted.

6.13.11 Scram from High Superheat Temperature

Outlet steam temperatures will be measured in the various steam channels. An excessively high temperature will cause scram since it indicates insufficient cooling which can cause fuel element failure. The signal for steam over-temperature scram will come from two sets of sensors. The first set consists of the thermocouples placed in the outlet of each fast core fuel assembly. The second set is a group of thermocouples located away from the reactor vessel in the main steam line where they will be accessible for replacement. They will be the primary scram sensors, since the in-core thermocouples can be replaced only when the fuel is replaced.

6.13.12 Scram from High Ratio of Flux to Steam Flow

This anticipates a dangerous rise in temperature and therefore would precede a high-temperature scram.

6.14 Manual Rod Control System

The reactor has 44 hydraulically driven and 20 twin-screw driven control rods, each of which is controlled manually from the control room. Selection of the one rod to be moved with manual control is accomplished by the use of two master selector switches. One switch selects the row array, and the second switch selects the column array within the electrical circuit corresponding to the "row" and "column" arrays in the control rod pattern. Thus, only the control rod located at the intersection of the "row" and "column" arrays is responsive to manual position control.

A pilot light, mounted on the position indicator for the selected rod, is energized to indicate which rod is responsive to manual positioning.

Actual rod control movement is accomplished by use of a separate switch. This switch energizes the solenoid valves in the hydraulic system and moves the selected rod to a new "Insert" or "Withdrawal" position.

6.15 Rod Position Indication System

The system provides simultaneous indication of each of the 64 control rods. Forty-four of the rods are of the locking-piston type and provide digital indication of the position of each; the other 20 rods are of the twin-screw design and provide self-synchronous indication.

Position indicators are provided to indicate positions of the rods throughout their full stroke, while pilot lights indicate which of the rods are at the "all in" or "all out" position.

The position indicators and fixed position pilot lights for each rod are grouped together and arranged on the control room panel in a pattern simulating the relative locations of the rods in the core.

6.16 Radiation Monitor System

6.16.1 Reactor Radiation Monitors

Fuel Element Rupture Detection: A fuel element failure in either fast or thermal core is first detected by monitoring the off-gas from the main condenser air ejector with gamma spectrometry instrumentation. By selective discrimination of the isotopes of the fission product gases, the sensitivity of the detection system is increased above that determined by the gross activity increase during fuel element failure.

Gamma spectrum analysis is provided to determine the extent of the failure. The location of the ruptured fuel element is determined as described in Section 4.9.

Two channels of instrumentation monitor the off-gas and serve two functions:

1. To detect a fuel element failure, and
2. To monitor off-gas activity and automatically close the off-gas line to stack in the event of excess activity in the off-gas.

6.17 Process Radiation Monitors

Instrumentation is selected for continuous monitoring of the radioactivity of certain processes. Critical processes, significantly high in radioactivity, are monitored for variation from normal.

6.17.1 Stack Gas Monitor

An isokinetic probe is employed to collect a representative continuous flow sample of the gas discharged from the stack. A fixed filter is provided to collect particulate matter from the sample stream for laboratory analysis of contamination. The stack gas sample is monitored by a dual function, gamma spectrometer for gross gamma activity, and for the activity of a particular isotope, such as Argon -41 or Nitrogen -13. Both measurements are continuously recorded to show trends and activity levels.

The stack gas monitor would be modified to provide a backup gross gamma monitor for use in the event that the dual function gamma spectrometer becomes inoperative for an extended period of time.

6.17.2 Liquid Process Monitors

The following liquid processes are monitored continuously and recorded intermittently to determine radioactive changes and trends.

1. Reactor Enclosure Cooling Water
2. Service Water Discharge
3. Circulating Water Discharge
4. Waste Demineralizer Effluent
5. Condensate Demineralizer Influent
6. Condensate Demineralizer Effluent
7. Reactor Cleanup Demineralizer Influent
8. Reactor Cleanup Demineralizer Effluent

Where required, continuous flow samples of these processes are taken so that hold-up time may be employed to allow short half-life isotopes to decay to negligible levels. In-line detectors may be employed where holdup is not required.

6.18

Area Monitoring System

Approximately 30 gamma-sensitive detectors are located throughout the plant, each arranged to monitor a specific area. The gamma flux level at each detector is indicated continuously in the station control room. Adjustable trips provide annunciation in the event of a high gamma flux level of any particular detector. Two multi-point recorders permanently record the gamma flux levels on a continuous cycle basis.

FIGURE 6.3.1.
BLOCK DIAGRAM FOR BASE LOAD PLANT OPERATION

FIGURE 6.3.2
PRESSURE REGULATOR SCHEMATIC

FIGURE 6.11.1

FIGURE 6.12.1

FIGURE 6.13.1

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7.0 SPECIAL TEST INSTRUMENTATION

Conventional boiling water reactors have exhibited adequately stable behavior over the required range of operating conditions. The Mixed Spectrum Superheater, because of its more complex nature, will introduce some new problems in reactor dynamics and nuclear performance. It is, therefore, desirable to provide investigation tools that will more readily yield data from transient phenomena or continuously monitor steady-state operation. In this section, special test instrumentations are described that will enable the reactor designers to investigate various thermal hydraulic and nuclear phenomena with a minimum of interference with plant operation.

7.1 Instrumented Fuel Assemblies

Instrumented fuel assemblies can be provided for investigating transient and steady-state reactor hydraulic performance. These assemblies will be designed so that they can be inserted in the core in place of any regular fuel bundle, to provide maximum flexibility for investigative purposes. For the present design, it is planned to place instrumented fuel assemblies in the thermal core, the thermal buffer and the fast core, making a total of three.

7.1.1. Thermal Boiler Instrumented Fuel Assemblies

Many instrument sensor groupings can be provided. Two arrangements to be used in the boiler(thermal core and buffer) are identified here, with the features noted in the table below:

<u>Instrumentation</u>	<u>Type I</u>	<u>Type II</u>
In-Core Fission Chambers	3	3
Calibration Tube	1	1
Thermocouples	9	9
Pressure Taps	3	3
Flow Meters	2	1

The Type I assembly will have both inlet and outlet flowmeters, while Type II will feature only the inlet flowmeter. All other features of the two instrumented assemblies will be identical.

All electrical sensor leads within the reactor vessel are mineral insulated with stainless steel sheathing. These leads, together

with the pressure tap lines and neutron flux monitor wire calibration guide tube, are gathered at the top of the instrumented assembly in a flexible protective stainless steel sheath. This sheath groups the leads for convenience and protects them from mechanical damage. The leads terminate at a disconnect flange within the reactor vessel.

Connection or disconnection can be made at the flange after shutdown of the reactor and removal of the vessel head. The water level would be lowered just below the flange and work performed from a thick steel plate which serves as a platform and gamma shield. The column of water above the core in the chimney and upper plenum, plus the shielding provided by the work platform, will attenuate radiation to sufficiently low levels that connection, testing and disconnection of the flanges can be made within permitted personnel exposure levels. Upon disconnection, a cover plate must be placed over both the vessel flange and the instrument lead flange before the vessel water level is raised or the vessel head replaced. A connecting cable bundle joins the leads at the internal vessel flange junction to a panel beyond the biological shielding. Junction points are provided at the panel for all of the thermocouples, the flow meter leads and the in-core ion chamber leads. The pressure tap lines terminate at a valve rack in the panel. This rack contains remotely operated valves to switch the pressure tap connecting tubes between differential pressure transducers or coolant backflow equipment. The transducers convert the pressure signal into an electrical signal for transmission to the control room. The backflow equipment provides a continuous flow of sub-cooled feedwater back through the pressure tap lines into the reactor vessel, until the moment of measurement. The function of this purge flow is to prevent vapor formation within the lines. The valves are operable from the control room.

All instrumented assembly measurements will be continuously recorded in the control room during operation. For the Type II assembly this will consist of:

9 Temperature Measurements

3 Differential Pressure Measurements

1 Flow Measurement

3 Neutron Flux Measurements

In addition, the Type I assembly will provide two additional flow signals; the outlet flow and the ratio of outlet to inlet flow for the investigation of two-phase fluid flow phenomena such as relative steam water velocity.

A great variety of measurements can be made. The following list identifies some of the information which sensor data will yield directly or permit continuous monitoring for investigative purposes.

1. Integrated fuel assembly power generation.
2. Integrated fuel burnup (U-235 depletion).
3. Transient power profile.
4. Steam generation rate in pounds per hour.
5. Coolant flow rate.
6. Subcooled inlet temperature.
7. Two-phase outlet temperature.
8. Two-phase outlet flow signal (Representative of coolant steam quality and void fraction).
9. Instantaneous thermal neutron flux density at three points.
10. Full length flux profile.

Continuous recording by strip chart recorders provides permanent records for both steady state and transient investigations. The fine transient response of the neutron flux monitors, the flow sensors and the differential pressure measurement equipment permit extensive transient performance analysis.

Typical transient phenomena which the instrumented assemblies and recording equipment would permit investigating are:

1. Reactor hydraulic and neutron flux response to rod oscillation and/or pressure change tests.
2. Hydraulic instabilities.
3. Coupled void-neutron flux reactions.
4. Fuel and reactor time constants.

While industrial strip chart recorders are usually adequate for most investigations, a special remotely initiated step change of chart speed could be provided on all pen recorders for greatly expanding the chart time scale. This will permit a more careful evaluation of transient data. In the event that even greater time expansion should be desired, a considerable variety of flexible high speed recorders can be used. See Figure 7.1.1.

7.1.2. Superheater Instrumented Fuel Assembly

As mentioned above, it is presently possible to obtain accurate measurements of temperature and flow within a boiling water core at saturated steam conditions with the use of turbine-type, mass-velocity flow sensors and resistance-thermometer detectors. These instruments comprise part of an Instrumented Fuel Assembly, and have proven successful under operating conditions at General Electric's Vallecitos Boiling Water Reactor.

a development program is presently underway to investigate behavior of these instruments under superheat conditions - temperatures of 1000°F and pressures up to 1000 psi. Regarding the resistance-temperature detectors, the superheat temperature can be measured, but high neutron and gamma flux levels complicate performance and shorten the detector operating life. Current studies of protective materials show promise that the detectors will operate successfully under superheat conditions.

Another development program involves a pneumatic temperature monitor system. This is based upon the principle of Boyle's Law relating to pressures and temperatures, where it is possible to measure the pressure of the superheat outlet plenum and compute the temperature at the same point. Measurement of the total temperature of the coolant gas is made by flowing some of the gas through two nozzles in series, and then measuring the ratio between the inlet pressures of the two nozzles. This ratio and the known temperature (T_0) are used to calculate the unknown temperature (T_1).

This approach permits design of a system which provides the required sensing accuracy without the life limitations of thermocouples. Also, there will be no replacement or repair of sensors or other devices within the core. Present studies have been based on a gas-cooled reactor where T_1 could be in the range of 1000-1200°F and T_2 approximately 100°F. However, superheated steam, in order to retain its near-true gas properties, cannot be cooled below 700°F. This is a much smaller margin than for helium, and complicates the question of accuracy.

Another phase of this development program is centered on the effectiveness of the pneumatic temperature monitor system in a superheated steam environment.

These development programs are now being carried out. The provision of the instrumented fuel assembly in the fast core will depend upon the successful conclusion of these tasks.

7.2 Boiler Downcomer Instrumentation

Thermocouples located in the reactor downcomer and lower plenum permit the investigation of feedwater mixing and steam carryunder phenomena by measurement of the fluid temperatures. Any number of thermocouples can be provided. Twelve have been arbitrarily identified for this system. The thermocouple leads would be mineral insulated with stainless steel sheathing. These leads would be firmly supported within the reactor vessel to prevent coolant flow-induced vibrational fatigue problems. In the upper plenum, the leads would be bunched in a cable for termination at an internal flange similar to that employed for the instrumented fuel assembly lead termination. A connecting cable would penetrate the dry well and terminate at a connection panel, again in a

manner similar to that used for the instrumented fuel assembly leads. All temperature measurements would be recorded on a multipoint recorder in the control room.

7.3 Traversing In-Core Probe

As an alternate to the fixed position In-Core Monitoring System, a Traversing In-Core Probe (TIP) may be used.

The Traversing In-Core Probe would utilize a miniature fission chamber which may be traversed over the full core length in any of the eight flux monitor thimbles located in the reactor core. The chamber may be positioned in various locations within any one of the eight guide tubes.

The TIP would provide instantaneous readout of the neutron flux at any point or, with a traverse, provide a continuous flux profile.

The system consists of the fission chamber affixed to a triaxial cable. A drive cable will be wound concentrically around the triaxial cable. The drive cable will have an outer winding spaced in such a manner as to form a fixed pitch helix along the cable. The motor driven hobbled wheel will engage the helix and provide positive movement in both the "in" and "out" directions with respect to the core. A position indicator will be included in the drive unit to determine the position of the chamber within any reactor channel. The triaxial cable will transmit the fission chamber signal to a flux amplifier. This instrument furnishes the high voltage supply to the chamber and the necessary amplification for visual display and recording readout.

The cable and chamber will be arranged to allow calibration of the chamber using a portion of the cable itself as a flux probe. A section of the cable adjacent to the chamber will be counted to establish the induced activity. Using standard counting techniques, the activity of the cable and the actual neutron flux at the in-core calibration point may be correlated. This will permit calibration of the chamber.

The traversing in-core probes would be mounted in a battery of eight such that any one or all eight TIP's may make a core traverse at any time. Individual controls would provide for positioning of the TIP's in their respective channels. By ganging the probes, a traverse of the core will result in a simultaneous flux plot of the entire core. The addition of recorders to the system will provide a complete record of the traverse.

To measure the axial flux variation in the fast core, at least one of the probes will be placed in the thermal buffer adjacent to the fast core shroud. Because the fast core is relatively small and tightly-coupled, it is believed that the fast flux leaking through the buffer would provide a measure of the axial flux distribution in the fast core. The radial flux distribution will be checked by the temperature measurements at the exit of each fast core fuel assembly.

7.4 Control Rod Oscillation Tests

An investigation of reactor transient response characteristics is of considerable interest to both the reactor operator and the reactor designer. The tests employed to obtain this information can take many forms, most of which involve disturbing the reactor system and measuring its response to the disturbance. A good means of providing a controlled disturbance is by means of oscillatory movement of a control rod. Performing such oscillatory movement requires modification of one or more of the conventional control drives and the addition of special equipment not normally provided for operation of a nuclear power plant.

Some of the reactor response characteristics, and thus the transfer functions, will tend to change with fuel burnup due to depletion of U-235 and production of plutonium. Thus, it may be desirable to perform rod oscillation tests periodically during core life to determine the degree of change. As the core burns up a greater depletion of U-235 and production of plutonium will occur in the center than in the outer regions. While this radial change of fuel characteristics is not expected to drastically change the system transfer functions, a partial fuel reload of a different fuel design or a different fuel material can be expected to have an appreciable effect. Accordingly, it may be desirable to investigate the reactor system transfer functions at any time during plant life.

The design of a control rod oscillation system has been approached with a twofold objective. First, that during normal reactor operation, all of the control rods have the same physical operating characteristics. Second, the objective is to provide a system which would require a minimum of reactor down-time and adaptation-time to place the reactor in test readiness or return it to normal operation. To meet these objectives, the approach has been to find ways of building the test capabilities into an existing control rod drive system so that they can be switched in or out as desired. Present investigations strongly indicate the feasibility of this approach.

FIGURE 7.1.1

INSTRUMENTED FUEL ASSEMBLY

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8.0 PLANT OPERATION AND MAINTENANCE

8.1 Introduction

This section describes the startup, shutdown, load following, and re-fueling procedures as they are presently envisioned for the Mixed Spectrum Superheating Reactor. In view of the more complex nature of the MSSR with its multiple zones of widely varying neutron spectrum, it is anticipated that its operation will, at least initially, be more difficult than will be the case for a conventional BWR.

The present design is intended to serve as a prototype for the Mixed Spectrum concept. As operating experience accumulates, it can be expected that the procedures discussed here will be confirmed or modified. The following discussion should, therefore, be considered to be preliminary. Its purpose is to indicate that, based upon present knowledge, sufficient provisions have been factored into the design of the plant to establish a feasible procedure for bringing the Mixed Spectrum Superheater into operation as a power producing reactor.

8.2 Basic Philosophy of Operation

Although the plant will be instrumented so as to prevent damage due to foreseeable accidents and maloperation, the experimental nature of the proposed plant makes it imperative that strict procedural control be exercised at all times. Basic operating principles will be established and operation will take place under a set of written detailed procedures aimed at preventing compromising of plant safety.

8.3 Preoperational Testing

A program of preoperational testing will take place prior to initial operation of the plant. The objective of this program will be to demonstrate that the plant has been built according to specifications and that it is ready for initial fuel loading and startup.

The preoperational test program will include all checks, adjustments, calibrations, and operations necessary to assure that initial loading and subsequent operation can be safely undertaken. Each system will be functionally tested with emphasis on systems important to the safety and operability of the unit.

The details of the testing program will be worked out as the design of the plant progresses.

8.4 Initial Core Loading and Startup

The initial startup and operating period will be carried out as a part of the Mixed Spectrum Superheat development program. It will consist of initial core loading, critical testing, and extended power operation

testing. It is expected that approximately one year will be required for this phase of the program.

Accurate knowledge of reactor core parameters is required for safe operation of the plant. A program of tests and measurements to verify design physics calculations will take place during and immediately following the initial fuel loading.

In general, the nuclear test program will explore the important nuclear characteristics of the core and determine that they are in accordance with the design calculations. If discrepancies are discovered, the effect of these on reactor operation and safety will be determined before operation of the reactor begins.

In this zero power testing period, the excess reactivity and shutdown margins in the core will be established and it will be determined that safeguard limitations have been met. The moderator temperature and void coefficients in the boiling core and the flooding coefficients in the fast core will be obtained. The critical control rod configurations for uniform and non-uniform control rod withdrawal will be determined. Finally, the detailed control rod patterns for the start-up will be established.

The details of the power test program have not yet been formulated. It is expected that a number of low power tests and tests at high steam flow and minimum superheat power will be made before the superheater is operated at high power and the superheat elements become highly radioactive. Assuming that the results of these tests are satisfactory, it is expected that the approach to full power will generally follow the procedure in Section 8.5.1 below.

8.5 Normal Operation

This section presents an outline of the proposed operating procedures for normal startup, shutdown and load following. The normal operating characteristics of the plant will be established by the power operation test program following the critical tests. Appropriate changes in these procedures will be made in accordance with the results of the reactor design development program and on the experience derived from the power operation test program. In particular, the startup and shutdown procedures are tentative and must be supported by heat-transfer tests and reactor operating experience before feasibility is assured.

The procedures proposed here are based, in part, upon experience derived from operation of the Dresden nuclear power plant. The departure from present technology centers, of course, in the control and cooling of the superheater section.

8.5.1 Cold Startup

A cold startup will occur during the initial startup and each time the reactor is returned to service following refueling or a maintenance shutdown. Initially, it is assumed that the reactor is cold, and that the following conditions apply:

1. System is at low pressure.
2. The reactor (both cores) is flooded.
3. The shutdown cooling system is on.
4. Turbine admission valves are closed, by-pass valves partially open.
5. All control elements are in their least reactive positions.

An outline of the procedures proposed for startup is given in the following paragraphs:

1. A complete and detailed startup list will be followed prior to beginning the actual startup to assure that all applicable equipment and systems, particularly safety circuits and scram mechanisms, are operable. Containment must be secured.
2. The turbine is taken off turning gear and the shutdown cooling system is turned off.
3. The superheater spray system and the superheater drain pump are turned on. The spray and drain pump are sized so that the net effect is to drain the fast core while safe temperatures are maintained by the spray. The unflooding coefficient may be positive, so that the drain line will be orificed to limit reactivity inputs to safe values.
4. Vacuum is drawn on the reactor by using a mechanical pump on condenser. The boiling temperature will be reduced to about 135°F so that boiling occurs quickly to supply steam cooling to the fast core. The steam is by-passed to the condenser.
5. If decay heat is not available to heat the boiler, the reactor is made critical by withdrawing the control rods (poison) in the outer region of the boiler.

6. When steam flow has been established, the fast core spray is turned off. Fast core temperatures are dependent upon the power split between the fast and thermal cores, as discussed in Section 8.6.4. The drain pumps are turned off when the drain hot-well runs dry.
7. The control rods in the thermal core and thermal buffer are withdrawn in a planned sequence.
8. With steam flowing, the reactor pressure is allowed to build up. Sufficient steam is dumped through the by-pass valves to adequately cool the fast core. Pressure build-up is controlled to maintain a temperature rise rate of 100°F per hour.
9. At approximately 150 psig, the pressure regulator is placed in service. The pressure build-up rate is then controlled by moving the pressure set point.
10. When the pressure builds up to about 300-400 psig, the steam jet ejector system is placed in service, and operating condenser vacuum is established.
11. When rated pressure and temperature (1000 psig and 545°F) are reached, turbine rolling and heating begins. Prior to this, the entire system has been heated by bleeding small amounts of steam throughout the system.
12. The steam superheat is raised slowly by increasing the power generated in the fast core. This is done in the following manner: First, the coupling control elements next to the fast core are removed. Then, the control fuel elements in the fast core are inserted slightly. Because of the nuclear coupling between the fast and thermal cores, this will tend to increase power in the boiler. The neutron flux in the boiler is, therefore, monitored and control rods are inserted, if necessary, to prevent excessive power generation in the boiler. This process is repeated until steam conditions approach the desired condition. The control rods in the thermal buffer will play a minor role, and it is expected that they will be used for final trimming purposes only.
13. The turbine is brought up to speed, and the generator synchronized and placed in the system at about 5 MWe.
14. Loading then proceeds based on schedule or system demand.

During the entire approach to power, the temperatures at the outlet of the superheat fuel elements are monitored and period protection is available in both the boiling and superheating section

The startup procedure outlined above appears to be practicable. Some technical difficulties remain to be resolved, but they do not appear to be insurmountable. The primary difficulty will lie in making the transition from spray to steam cooling. There will be a period of time when the fast core is subjected to both spray and steam. The pressure drop and cooling capabilities under these conditions must be established by heat-transfer tests to determine the point at which the spray can be turned off.

Instrumentation will be provided to indicate the steam conditions and flow rates for the low mass flow rates during the initial low-pressure part of the startup. This may take the form of a small flowmeter, either in the by-pass line or line paralleling the by-pass, in addition to the temperature and pressure indicators used for normal plant operation. Conventional components can be used, and it appears that an adequate indication system can be designed without significant development effort.

8.5.2 Hot Startup

If the plant is shut down for a short period of time, it is maintained in a hot, stand-by condition. The reactor vessel and auxiliaries remain at or near operating temperatures and steam flow is maintained at a low level and by-passed to the condenser.

The hot startup procedure is essentially independent of the cause of shutdown provided that the cause is recognized and corrected. The reactor instrumentation is reset and downscaled where required. A hot startup check list is completed prior to any movement of control rods. A check is made to assure that all control rods are coupled that had scrambled since the prior coupling check.

To start up, critical rod positions are estimated based on the critical rod positions for the previous run with an allowance for xenon build-up. Startup then proceeds according to steps (7) through (14) of the cold startup procedures. It is not necessary to use the fast core spray cooling system since the fast core is adequately cooled by the steam generated by the decay heat in the boiler. This aspect is discussed in more detail in the section on Short Duration Shutdown.

8.5.3 Reactor Control Under Load Changes

Conceptually, the normal load following procedures for the present plant will be the same as those described for the 300 MWe reference design (GEAP 3590). This assumes that load changes are pre-scheduled or are initiated by communication from the control load dispatcher. Sudden unexpected decreases in loads are handled by by-passing part of the steam flow until the reactor power can be trimmed back to the point where the by-pass valves can be closed

without overspeeding the turbine. Sudden large increases in load demand are not anticipated; however, if they should occur, the pressure regulator will maintain reactor conditions unless the load change is severe enough to cause a scram.

For a scheduled load (steam flow) increase, the power in the boiling core will be increased a small amount by withdrawing control rods in that region. The nuclear coupling will cause the power in the fast core to rise also, but not as much as will be required to maintain the required power split. The superheated steam temperature will therefore drop. If the drop in temperature is large enough (about 25°F), the power in the superheat region would be raised by moving control rods in that region and trimming the thermal control rods.

A reversed procedure is used for a load decrease. The superheater core power is reduced enough to decrease the outlet steam temperature about 25°F. The steam flow is then reduced by decreasing the boiling region power to the point where the steam temperature is raised back to the set value. The steps continue until the desired flow and temperature are achieved. Automation is not considered for operating the prototype, but it is expected that for normal power production, operational procedures can be automated once the plant characteristics have been established.

8.5.4 Extended Shutdown

A normal shutdown of long duration will be accomplished as follows:

1. The main generator load is reduced, and the turbine-generator is separated from the system. The fast core power is simultaneously reduced by withdrawing the control fuel elements and inserting the coupling control poison blades between the fast and thermal cores.
2. The control rods in the thermal boiler are withdrawn as required to maintain power in the thermal core. Calculations show that the thermal core power can be maintained at about 90% and the fast core at below 10% of normal full power operation.
3. When the fast core and steam temperatures reach equilibrium, the superheater spray pump is turned on to spray saturated water on the fast core. The maximum fuel surface temperature at this time is estimated to be about 600°F so that spraying saturated (540°F) water on the fuel will not result in significant thermal shock.
4. The thermal boiler power can now be reduced as required by manipulating the control rods. The superheater drain pump is

turned on to prevent premature flooding of the fast core. Pressure is maintained with the pressure regulator.

5. All control rods are inserted, and control rod withdrawal circuits are locked.
6. The reactor pressure is reduced by controlled steam flow to the main condenser through the by-pass valves. This steam is desuperheated, condensed, and returned to the reactor vessel by the reactor feed pumps. The maximum rate of cooling of the reactor is 150°F per hour. The steam by-pass valves are closed when the reactor pressure drops to about 120 psig.
7. The superheater drain pumps are shut off, and the fast core is allowed to be flooded. The superheater spray pump is shut off when fast core is completely flooded. A differential pressure indicator which measures the height of water in the fast core is used to determine when flooding is complete.
8. The shutdown cooling system is placed in operation. This system cools and maintains the reactor water at 120°F.

8.5.5 Short Duration Shutdown

A normal shutdown of short duration may be accomplished while maintaining system pressure. The turbine-generator is unloaded and separated from the system. Reactor decay heat is removed by by-passing steam to the main condenser or to the emergency condenser if the main condenser is not available.

Assurance that the steam generated by the decay heat in the thermal core can adequately cool the fast core is provided by maintaining the feedwater temperature at its normal level. This is done by the system sketched in Figure 8.5.1. Lines are provided between the turbine by-pass line and the feedwater heaters to utilize part of the steam to heat the feedwater. Check valves are provided as shown to prevent back-flow into the turbine or condenser.

The system shown in Figure 8.5.1 can prevent overheating in the fast core provided the reduction with time of the decay heat in the fast core is not significantly less than that in the thermal core. The heat transfer coefficient varies approximately as the 0.8 power of the mass flow rate. The film drop is directly proportional to the heat flux and, therefore, to the power generated; the bulk coolant temperature rise remains constant. It can be seen that if the total reactor power is reduced while maintaining the power split constant, the net effect is to lower the fast core fuel temperatures. Thus, the reactor will always be adequately cooled for a power reduction, including scrams, if:

1. The feedwater temperature is not reduced significantly.
2. The relative power split between fast and thermal cores is maintained such that the power produced in the fast core is always less than 25% of the total.

It appears that both of these requirements can be satisfied under normal operating conditions.

In case any difficulty should be encountered in designing the system as described above, an alternate scheme has been considered to provide the necessary steam flow. In the previous case, attempts are made to maintain both reactor pressure and feedwater temperature at their normal levels. The alternate plan is to override the pressure regulator and maintain sufficient steam flow by lowering reactor pressure and flashing steam. The rate of pressure decrease would be regulated by measuring the steam temperature at the superheater outlet and regulating flow to keep the steam temperature below a predetermined level. This plan is feasible but has the disadvantage that the reactor must be brought back up to operating pressure to get back on the line.

8.5.6 Cooling During and Following Scram

As discussed in the previous section, the thermal core is adequately cooled by boiling following a scram. The fast core is prevented from overheating by the resulting steam flow since the feedwater temperature is maintained either by the extraction flow in the turbine or through the by-pass in the main turbine by-pass line if the steam is by-passing the turbine.

If the cause for scram should also require containment building isolation, the emergency condenser is utilized for cooling. If steam flow becomes inadequate, the spray cooling system is used.

8.5.7 Emergency Shutdown

Large load rejections will not necessarily result in scrambling the reactor since the turbine by-pass valve opening will be initiated by a signal from the generator circuit breaker. The turbine admission valves would then be closed to prevent overspeeding the turbine.

Sudden closure of the turbine admission valves, however, would cause the reactor to scram on high flux or high pressure. If this occurs, the by-pass valve opens at its over-pressure set point to maintain pressure by shunting excess steam to the condenser.

If the turbine by-pass valves fail to open, an emergency condition exists, and the emergency cooling system must be put into operation. Calculations show that at full power with no cooling, the emergency

cooling system must be put into operation within three seconds to prevent fuel melting in the superheating section. With the reactor scrammed, the time available is considerably longer. Since all valves can be fully opened in less than one second, it appears that requirements placed on the emergency cooling system can be met.

8.6 Normal Refueling

Refueling of the Mixed Spectrum Superheater will be carried out on a batch basis with a portion of the fuel being replaced each time. Fuel cycle management studies with proposed refueling schedules are discussed elsewhere in this report. These studies will be used as guides for estimating fuel costs. For the equilibrium fuel cycle in both superheater and boiler, the calculated refueling schedule calls for refueling every 11.5 months. For actual operation, burnup data for individual fuel assemblies will be obtained from power level records, from the in-core monitoring system, and from periodic gamma probing of fuel assemblies during shutdown. These data dictate when a given fuel assembly should be replaced. The specific date chosen will depend upon system requirements and the status of other generating units. Shutdown and cooling are carried out in the normal manner, and removal of the vessel head and reactor internals above the core follow as soon as possible. The refueling extension tank above the reactor is filled with water, and refueling can proceed. The procedures for refueling the thermal and fast cores are described separately below.

8.6.1 Refueling the Thermal Core

The procedures used to refuel the reactor are outlined as follows:

1. New fuel is transported to the fuel storage pool by reactor crane sometime prior to actual reactor refueling operation.
2. The top-mounted control rod drives (4 fast core and 16 thermal buffer controls) are detached from their respective control elements with the control elements in their least reactive positions.
3. The vessel top head is unbolted, removed and stored in the location provided for it.
4. The refueling well above the reactor is filled with water, and the door to the fuel storage pool is opened. The water level will be approximately 24 feet above the vessel parting line.
5. The steam dryer is removed in four segments and stored in the fuel storage pool.
6. The steam downcomer is unbolted, lifted out of the reactor and stored in the fuel storage pool.

7. With the refueling equipment described in Section 4.11.1, unloading, fuel shuffling, and loading are carried out as follows:

a. Unloading:

Remove fuel assembly hold-down.

Align refueling platform over fuel assembly.

Grapple fuel assembly in core with fuel grapple.

Move fuel assembly through slot to fuel storage pool.

Align refueling platform over storage rack.

Lower fuel assembly into fuel storage rack.

b. Fuel Shuffling:

Grapple fuel assembly in core with fuel grapple.

Lift fuel assembly with grapple hoist until lower end clears chimney.

Move fuel assembly to new position previously vacated according to unloading procedure.

Lower fuel to new position in core.

c. Loading:

Align refueling platform over storage rack.

Grapple new fuel assembly in storage rack.

Move new assembly through slot to reactor well.

Align refueling platform over proper core location.

Lower new fuel assembly into reactor.

Secure fuel assembly hold-down.

8.6.2 Refueling the Fast Core

The fuel handling procedures used to refuel the fast core are identical in concept to that used for the thermal core. Fuel handling is made somewhat more difficult by the closer spacing of the fuel elements. Also, the flange used to bolt and seal the steam downcomer to the

fast core shroud limits access to the fast core. This flange prevents the operator from positioning the refueling platform directly over the fast buffer and outer row of fast core fuel elements so that an offset grapple head is required for these fuel elements. This arrangement also makes it necessary to remove fuel elements from the central portion of the fast core before either the fast buffer or outer row of fast core fuel elements can be removed. The operational inconvenience resulting from the present flange design indicates that further work in this area is desirable. It does not appear that feasibility of the concept will be affected.

The present fast core design allows a maximum clearance between adjacent fuel elements of about 0.100 to 0.200 inch. The individual fuel elements are held loosely in place by inserting the lower end of the fuel elements into the lower grid plate. Sufficient spacing is provided between the nose pieces and the grid so that the fuel elements can adjust to the inward movement of the flexible fast core shroud.

The loading of the fast core fuel must be carried out through a water depth of about 40 feet. Some difficulty can be anticipated in inserting the fast core fuel assemblies into the fast core because of their close clearances. To facilitate insertion of the fuel assemblies, a special spreading device (Dwg. No. 762D733, Fig. 8.6.1) has been designed to assure a maximum opening. The spreading device is inserted into the fuel opening, and pressure is applied to the adjacent fuel assemblies. A shear pin is provided in the jacking mechanism to prevent application of forces that could damage the fuel assemblies. The maximum fuel movement will be of the order of 0.100 inch so that reactivity effects due to fuel movement are expected to be negligible.

With the special devices described above and with practice, the reactor operators should be able to refuel the fast core with only slightly greater difficulty than would normally be encountered in refueling the boiling water portion of the reactor. Mock-ups and dry runs during the design development stages are planned to reveal areas where improvement can be made in the present design.

Part of the refueling procedure involves measuring the reactivity of the unflooded fast core to insure that, first, no elements are too highly enriched and, second, that at the completion of reloading, the excess reactivity in the cold unflooded condition is approximately equal to the 2.1% design value (cf. Table 3.5.3). An approximate guide on the worth of each assembly can be obtained from calculating subcritical multiplication in the flooded condition after each new assembly is loaded. During the first core

loading both wet and dry critical measurements can be made. After the fuel has operated at power and is generating significant decay heat, measurements in the unflooded condition require cooling for the fast core. The water from the spray cooling may obscure these measurements so that a different cooling source may be required. One possibility is to place the head on the vessel and perform the measurements under vacuum as described in steps 4. to 6. of the startup procedure described in 8.5.1. If many measurements are required this is apt to be a cumbersome and a temporary gas cooling arrangement may be necessary. The necessity and method of performing dry critical measurements on the fast core has not yet been studied in detail.

FIGURE 8.5.1 SKETCH SHOWING BY-PASS LINES ON MAIN TURBINE BY-PASS LINE TO MAINTAIN FEEDWATER TEMPERATURE DURING SHORT DURATION SHUTDOWN.

FIGURE 8.6.1

9.0 STEAM CYCLE

9.1 Preliminary Heat Balance

The flow diagram for the 75 MWe MSSR prototype is given in Drawing 985C846, Figure 9.1.1. The preliminary heat balance is given on Drawing 985C847, Figure 9.1.2. The estimated gross heat rate at rated power is 9380 Btu/kwh for a feedwater temperature of 300°F and turbine throttle conditions of 950 psia and 950°F. For the comparable saturated plant, the estimated heat rate is 11,150 Btu/kwh. The improvement in thermal to electrical conversion is therefore about 19%.

The most significant difference in the present heat balance diagram from that estimated for the 300 MWe plant is in the number of feedwater heaters. In the 75 MWe plant, only two feedwater heaters are provided. This is due to the fact that the plant incorporates a natural circulation boiler which requires low (about 300°F) feedwater temperatures. Further work to optimize the steam cycle may result in raising the feedwater temperature which would improve cycle efficiency.

10.0 TURBINE PLANT

The turbine-generator comprises a single-shaft, tandem compound, double flow 3,600 rpm turbine connected to a hydrogen-cooled generator with a direct-driven exciter. The unit is rated at a maximum capability of 75,000 gross kw with a primary throttle superheated steam flow of 575,000 lb/hr at 950 psia 950°F with operation at 1.5 in. Hg absolute back pressure and 0.5 per cent demineralized makeup.

Moisture separators are specified in the cross-over only; however, provisions will be included to add additional separators if it is later desired to operate with saturated steam. Turbine construction materials are considered to be conventional, but should be made compatible with the impurity content which is higher than for non-nuclear plants. The hydrogen and oxygen content will be higher, and the steam will also contain small quantities of Nitrogen-16 and Oxygen-19 as well as radioactive corrosion products from the reactor.

Pressure regulators are provided to operate the turbine control valves and by-pass valves to maintain automatically the system pressure. The turbine by-pass valves are sized to pass 100 per cent of the rated primary steam flow to provide for loss of electrical load. The by-pass valves will discharge into the main condenser.

The turbine shaft sealing system is designed to prevent the leakage of steam into the turbine room. The system is designed to seal the turbine with a manually-operated by-pass whenever the steam pressure to the regulator is 100 psig or greater. Sealing is automatic for pressures of 350 psig or greater. Provisions are made in the turbine shaft sealing system for draining water or other decontaminating fluid through the shaft seals and the shaft packing leak-off lines with no leakage to the atmosphere with the turbine on turning gear. Means are also provided for introducing decontaminating fluid into the turbine casing and indicating its level. Valve stem leak-off from all large valves, such as control valves, stop valves, and isolation valves, exhaust into the gland seal condenser. The gland seal is equipped with two blowers, one of which is a spare.

10.1 Condensing Plant

10.1.1 Condenser

The main condenser is a horizontal, two-pass type, located beneath the turbine exhaust and connected to it by an expansion joint. The cooling water enters the building from beneath the floor, passes through the water boxes, and leaves through pipes leading to the underground discharge canal. The condenser can maintain a back-pressure of 1.5 in. Hg absolute at rated turbine conditions.

The condenser is capable of handling, without undue pressure rise, a steam by-pass flow equal to 100 per cent of saturated throttle steam flow. A desuperheater is included in the turbine by-pass line.

The condenser also serves to remove gases from the condensate, acting as the sole deaerator of the system. A storage type hotwell is provided with sufficient capacity, based on 100 per cent of turbine throttle flow, to provide a 3-minute delay time for decay of short-lived radioisotopes in the condensate. Special care must be taken by the condenser manufacturer to provide a leak-tight connection between the condenser tubes, and it is expected that Admiralty metal tubes, welded to silicon bronze tube sheets, will provide the necessary protection against leakage.

10.1.2 Air Removal Equipment

Condenser air removal is accomplished by two, twin-element two-stage, steam-jet air ejector units with inter-and-after condensers, one of the units being standby. Primary steam is supplied to the air ejector through a pressure reducing station, with cooling water for the inter-and-after condensers provided by condensate from the condensate pump discharge line. The air ejector capacity is great enough to handle not only normal leakage, but also the free H_2 and O_2 resulting from coolant decomposition within the reactor. A motor driven mechanical vacuum pump is included for start-up operations.

10.1.3 Condensate Pumps

Two full-capacity, vertical, multistage, centrifugal, submerged suction condensate pumps are adjacent to the main condenser on the ground floor. These are provided to pump condensate from the condenser hotwell through the air-ejector and gland seal condensers, condensate demineralizer, regenerative heat exchanger and the two feedwater heaters to the reactor feed pumps suction header.

10.1.4 Circulating Water Pumps

Two, 60% capacity, vertical, circulating water pumps are located in the screen house. These pumps are a submerged suction propeller or mixed-flow type, with the driving motor mounted above the floor level.

10.1.5 Circulating Water System

Cooling water for the condenser, and for cooling of turbine oil coolers, generator H, coolers and turbine building closed cooling system, is drawn from the river at the screen house. This water flows through travelling screens which remove trash and debris.

Water is discharged from the circulating water pumps and flows to the station through two pipes below grade. Service water pumps discharge a smaller quantity of river water which flows to the turbine and reactor buildings for equipment cooling requirements. Circulating water discharges from the condenser through a concrete canal to a point to the river downstream from the screen house. Also located in the screen house are the fire pump, screen backwash pumps, and related equipment.

10.2 Feedwater System

This system includes that part of the circuit from the condenser hotwell to the reactor feed inlet. The condensate pumps take their suction from the condenser hotwell and discharge the condensate through full-capacity condensate demineralizers, through the steam jet air ejector condenser, gland seal steam condenser, the regenerative heater and the two bled-steam feedwater heaters, to the reactor feed pump suction.

10.2.1 Condensate Demineralizer System, Figure 10.2.1

The condensate Demineralizer System provides full flow purification of condensate to insure high purity reactor feedwater. It removes corrosion and fission products entering the steam system and eliminates the possibility of reactor feedwater contamination from condenser cooling water leakage. The number and physical size and cost of the mixed bed ion exchange units is minimized by high unit flow rates.

Specifications for maximum limits of effluent water from the condensate demineralizer are:

Conductivity, micromhos per cm at -4°C (25°F)	0.1
Iron, ppb	8
Copper, ppb	8
Cobalt, ppb	0

Cation resin is high capacity sulfonated polystyrene in the hydrogen ion form after regeneration. The anion resin is, Type I, strong base quaternary amine, in the hydroxyl form after regeneration.

The condensate Demineralizer System includes two mixed bed demineralizers and separate automatic regeneration equipment consisting of a cation resin regeneration tank, anion resin regeneration tank, a resin mixing and storage tank, and the necessary valves, instruments and controls. Transfer of resins between tanks is by

hydraulic means using air or water pressure or water ejectors for motive force. No significant attrition of resins results from this transfer.

The two demineralizer tanks operate in parallel and either is capable of processing the full condensate flow. The other unit is a standby unit and makes it possible to have one unit out of service for removal and replacement of resins. Each demineralizer tank is designed for a flow rate of 50 gallons per minute per square foot of bed area and does not exceed 6.5 ft. in diameter. Resin bed depth is 3 ft. minimum and the pressure drop, when spent, does not exceed 50 psi. A temperature alarm is included to prevent overheating the resins.

The system is operated with one of the demineralizers in service. When the resin bed is spent, as denoted by volume through-put, effluent conductivity and/or pressure drop, it is manually removed from the cycle and replaced in service by the standby unit. Transfer of resins to the cation resin regeneration tank is then manually initiated. When the demineralizer is empty, a spare regenerated charge of resin is transferred by manual initiation from the resin mixing and storage tank. After the short recycle wash, this unit is ready for service and becomes the standby unit. Regeneration cycle for the spent resins transferred to the cation resin regeneration tank, is then initiated and continues automatically. This cycle includes: separation of cation and anion resins, backwashing of resins, regeneration and rinse of resins, transfer of cation and anion resins to mixing and storage tank, final mixing and rinse so as to be ready for use.

The mixed bed units also act to filter out undissolved solids. During startup, substantial quantities of solid corrosion products are filtered out. These are readily removed by backwashing the resins in the regeneration system.

Because of the activity retained by the resins, the demineralizers are shielded. Valving and instruments are located outside the shield wall so as to be accessible for contact maintenance without need for removing demineralizer resins.

Regeneration equipment also requires shielding and is located in a separate compartment adjacent to the demineralizer tanks. This equipment will be accessible for contact maintenance during periods when it is not in use. Activity levels are nominal when resins have been regenerated and after disposal of regeneration solutions.

10.2.2 Feedwater Heaters

Two horizontal feedwater heaters of the shell and tube type with removable tube bundles are used. These heaters are located in the

auxiliary bay, with one on the ground floor and one overhead on a platform. The heaters utilize heat in the turbine extraction steam to heat the condensate at rated power from 103°F at the clean-up regenerative heater discharge to 300°F at the reactor feed pump. Drains from the heaters flash to the lower pressure heaters and ultimately to the condenser hotwell.

10.2.3 Reactor Feed Pumps

Two reactor feed pumps are provided, each for full load capacity. The pumps are horizontal barrel-type, multistage, with mechanical shaft seals, driven by fixed speed, 3000 V motors. The pumps are provided with automatic recirculation systems discharging to the condenser, to protect the pumps from over-heating at low loads.

10.2.4 Auxiliary Feed Provisions

In case of loss of auxiliary power, auxiliary feed to the reactor is provided by the control rod drive feed pumps. The normal supply to these pumps is the main condensate line from the condensate pumps. In an emergency, these pumps are supplied directly from the condensate storage tank.

10.3 Electric System

These are two major electrical systems in the station, the main power system and the auxiliary power system, Figure 10.3.1. The main power system consists of the main generator, as the source of power, and the connecting leads and transforming devices necessary to deliver the generated power to the high voltage transmission system. The auxiliary power system consists of the equipment necessary for the distribution of electrical power to all of the electrically powered devices and loads which are necessary to the operation of the plant under both normal and abnormal conditions, planned maintenance and repair, the comfort and well-being of plant personnel, and the safety of both operating personnel and the public in general.

10.3.1 Main Power System

Mechanical energy is converted to electrical energy in the main generator. The generator is a conventional three-phase, hydrogen-cooled machine rated at 60 cycles (3600 rpm), 93,800 KVA, 0.8 power factor. The 13,800 volt (line-to-line) terminals are connected by isolated phase bus duct to the main power transformer and to the station auxiliary power transformer. The main power transformer is an oil cooled, two winding, unit which transforms the generated power from the generator voltage level to 13,800 volts to the transmission system voltage level of 50,000 volts. Lightning arresters are mounted on the transformer to provide good lightning protection through close coupling to the high voltage terminals.

A direct-connected, shaft-driven exciter provides field excitation for the generator. A free standing metal-clad equipment enclosure houses the field circuit breaker, the voltage regulator and associated equipment.

10.3.2 Auxiliary Power System

Station auxiliary system electrical loads are supplied at several voltage leads. The larger motors, larger than approximately 150 horsepower, are supplied from a 4160 volt bus. This bus is supplied with power from one of two 6000 KVA power transformers. One of the transformers is directly connected to the main power system. The other transformer is connected to the 13,800 volt utility distribution system. The two transformers are physically and electrically interchangeable. All switchgear for the 4160 volt system is of the fully integrated, metal-clad type with completely enclosed and protected high voltage bus work and equipment. All necessary potential and current transformers for metering and relaying purposes, as well as all control circuitry and equipment, are included. The circuit breakers are removable and completely interchangeable.

Motors, smaller than approximately 150 horsepower, and other auxiliary electrical loads are served from a subsidiary 440 volt system. The system is wye connected to provide a ground for relaying purposes and to provide single phase service for general 220 volt use. Two low-voltage metal-clad switchgear assemblies constitute the main buses for this system, providing direct service to the larger motors and to the subsidiary motor control centers. These motor control centers are also of metal-clad, dead front, construction and provide control and circuit interrupting equipment for lesser loads. A diesel engine driven generator, to power essential auxiliaries in the event of loss of station power, is connected to either 440 volt bus at the operator's discretion. A load shedding switch permits the operator to quickly disconnect all non-essential loads from the 440 volt system to prevent overloading the generator. Loads considered to be essential include the following:

Auxiliary service water pump

Station battery charger

Rod drive seal pump

Instrument air compressor

Diesel oil transfer pump

Closed cooling water system pump

Core spray pump

Liquid poison pump

Critical instrumentation

In addition, the 220 volt station battery provides power for the DC turbine auxiliary oil pump, emergency lighting, instrumentation and DC control power. The emergency lighting is automatically switched on upon loss of normal AC power. An inverter transforms the direct current into alternating current for instrumentation usage. The battery normally floats across the 440 volt AC system on a trickle charge selenium rectifier.

FIGURE 9.1.1

UNLESS OTHERWISE SPECIFIED USE THE FOLLOWING:--				TOLERANCES ON MACHINED DIMENSIONS			TITLE	
APPLIED PRACTICES	SURFACES	FRACTIONS	DECIMALS	ANGLES			PROPOSITION	
✓	✓	+	+	+			HEAT BALANCE	
							FIRST MADE FOR MSS 75MW MCAP 3004-B	

985C 847

ESTIMATED GROSS PLANT HEAT RATE

$$\frac{206\text{ MW} \times 3413 \times 10^6 \text{ B/MW-HR}}{75000 \text{ KW}} = 9380 \text{ BTU/KW-HR}$$

FIGURE 9.1.2

FIGURE 10.2.1

FIGURE 10.3.1 ELECTRICAL ONE LINE DIAGRAM

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11.0 GENERAL SERVICES SYSTEM

11.1 Cooling Water System

The reactor services cooling water system is a closed loop system which supplies clean demineralized water at 75°F to the following components:

- Shield cooler
- Reactor enclosure drain tank
- Sample coolers
- Waste collector tank cooler
- Waste vapor condenser
- Fuel pool cooler
- Shutdown heat exchanger
- Shutdown pump cooler
- Emergency condensation tank cooler

Basic equipment for this closed loop system consists of two full-capacity heat exchangers, two full-capacity circulating pumps, and an expansion tank. The heat exchangers are cooled by 60°F river water supplied from the plant service water system. Makeup for the system is provided by the demineralized water transfer pumps.

11.2 Makeup System

To provide high purity water for reactor makeup and other uses, a make-up system is required. The water meets the following specifications:

Conductivity	1 micromho
Chloride	0.01 ppm
Silica	0.01 ppm

The system has enough capacity to supply initial water for filling the reactor, to meet fuel pool and refueling water requirements, and to handle any unusual makeup needs. It is estimated that the required net output of the system will be of the order of 10,000 gal/day. The make-up system and associated storage tanks will be in operation for a sufficient time before completion of the reactor plant to provide the water necessary for flushing the reactor and filling it for operation.

Components of the makeup system depend upon raw water characteristics of the specific reactor site; however, minimum equipment requirements include a cation exchanger, anion exchanger, and mixed bed exchanger. A filter and degassifier may be added, depending on water quality.

11.2.1 Demineralized Water Tank

The demineralized water tank is of aluminum construction and has a capacity of approximately 20,000 gallons. This tank provides storage for demineralized water from the makeup water system.

11.2.2 Condensate Storage Tank

The condensate storage tank is of carbon steel construction with a capacity of about 30,000 gallons. In addition to this tank which will serve the plant under normal operating conditions, an auxiliary storage tank with a capacity of about 65,000 gallons is provided. This auxiliary tank is located outdoors at grade level to supply such peak water demands as occurs, for example, during refueling.

11.3 Service and Instrument Air Systems

Two motor-driven air compressors with non-lubricated cylinders, one of which will be stand-by, will supply service air to the plant. Similarly, two such compressors of small size, one of which is stand-by, will supply air as required through moisture removal equipment to the instrument air system.

11.4 Heating and Ventilation

An auxiliary oil-fired boiler, complete with fuel oil storage and transfer system, is furnished for plant heating by means of steam heating coils located in the ventilating air supply ducts. The boiler will be located in the ventilation building located adjacent to the turbine building.

Separate ventilation systems, each consisting of two, full-duty fans located in the ventilation building and the associated air ducts for taking suction from the ventilated building and exhausting to the stack, are provided for:

1. The reactor building with an air flow of approximately 7800 scfm. This would result in several complete air changes per hour and be able to maintain the containment building pressure slightly (approximately 3 inches of water) below ambient.

2. The turbine building with an air flow of about 25,000 scfm.
3. The waste building with an air flow of about 4000 scfm. High-efficiency filters are located in the discharge between fan building and stack.

11.5 Fire Protection

Final provisions in this area are dependent upon the reactor site and the requirements of the reactor operators. The details of the fire protection system are expected to be more or less conventional and are left to be worked out during later stages of the project.

11.6 Domestic Water Supply

A domestic water supply system supplied probably from deep wells is required to furnish the drinking and sanitary water for the plant. This system includes pumps, treating equipment, chlorination equipment, and an accumulator tank.

11.7 Communication System

Telephone communication is provided between the central control room and the reactor operating floor. Telephone jacks are conveniently located for use during operation, refueling, and maintenance.

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12.0 SAFEGUARDS CONSIDERATIONS

12.1 Introduction

Safety requirements are a fundamental consideration in the design of each component of the present conceptual design. The basic safeguards philosophy guiding the design of the plant is summarized in this section.

Detailed safeguards analyses have not yet been carried out for this plant; however, a number of possible maloperations have been considered for which provisions have been made in the design of the plant. These considerations, including a preliminary estimate for the "Maximum Credible Accident", are discussed in this section to indicate the steps taken to meet the safeguards objectives.

12.2 Safeguards Objectives

The safeguards objectives guiding the design of the present plant are identical to those used for all power reactor systems. Design provisions for meeting these objectives, however, are more extensive because of the experimental nature of the plant. The general safety features to be achieved in this plant are briefly summarized as follows:

1. Normal operation of the plant must not result in the exposure of any persons on or off the plant premises to radiation in excess of the current recommendations of recognized national radiation protection groups.
2. Safety against a nuclear accident that might release dangerous amounts of radioactive materials must be preserved for all credible contingencies arising from equipment malfunction or operator errors.
3. Confinement of any significant quantity of radioactive materials that might be released from the reactor must be assured in the event that a serious accident does occur.

The provision of these features can be considered in three broad categories: Control over radiation exposure levels and release of radioactive material during normal operation, accident prevention, and absorption with minimum damage of the effects of possible but highly improbable accidents.

12.3 Normal Operation

Features important to the control of radiation exposure arising from normal operation include:

1. The design of gaseous waste handling so that quantities routinely

released to unrestricted areas are minimized. Provisions in the design for measurement and procedural or automatic control of the release based on measured results.

2. Location or shielding of radiation sources inherent in the system to the extent necessary to minimize personnel exposure during the performance of normal operating tasks.

12.4 Accident Prevention

Features important in preventing serious nuclear accidents include:

1. Provision of control rods in three separate regions of the core: the thermal core, the buffer region, and the fast core. If the fast core control bundles fail to scram, insertion of the coupling control rods between the two buffer regions will shut down the fast core.
2. A separate and independent shutdown mechanism in the form of a liquid poison injection system in the boiling core. The liquid poison system is manually operated and acts as a back-up to the mechanical control drive system.
3. Provision of systems for emergency removal of reactor heat. The primary heat removal system is the main condenser, which can be used to remove heat under many emergency conditions. If the reactor must be isolated from the condenser, other means are provided to remove reactor heat. These include: an emergency condenser located in the spent fuel storage pool, a core spray system, and a separate tank within the containment building to condense steam for a short time following scram if containment isolation is required.
4. The inherent safety of BWR's, that is, the negative void and temperature coefficients of reactivity and the strong Doppler coefficient of reactivity in the low-enriched, thermal core fuels.
5. The significant negative Doppler coefficient in the fast core. The coefficient is smaller in magnitude than for the thermal core; however, the available excess reactivity is smaller also.
6. Inclusion of epithermal poisons in the fast core fuel which causes a large negative reactivity reduction upon flooding of the superheating section. The reactivity reduction in the flooded condition is such that fast core criticality cannot be achieved even if all control elements on the

superheating section are removed.

7. Provision of a reactor safety system with sensing devices to detect and prevent potentially unsafe operating conditions from becoming severe.
8. The design of the reactor and primary equipment control system in such a way as to minimize the possibility for operator error or malfunction of equipment that could lead to an unsafe operating condition. Interlocks will be provided where necessary.

12.5 Accident Mitigation

By adhering strictly to the design safety philosophy outlined above and maintaining strong procedural control, it is felt that the probability of a serious nuclear accident can be made acceptably low for this plant. It is, however, a prototype facility whose characteristics will remain in doubt until demonstrated by actual reactor operation. To provide further assurance that the health and safety of the public is adequately protected against any unforeseeable occurrence, the following provisions are made:

1. Constant monitoring and automatically controlled isolation equipment on the primary system off-gas discharge line.
2. Housing the reactor and its principal auxiliaries within a vapor-tight containment vessel.

12.6 Safeguards Evaluation

The purpose of this section is to discuss those analyses that have been considered to show that a 75 MWe prototype for the Mixed Spectrum Superheater can be designed to be safe and that provisions can be made to prevent the occurrence of dangerous situations. Assurance that the plant meets the safety requirements will be provided during the detailed design stages when the design becomes more clearly defined. The preliminary safeguards considerations, as outlined here, form the bases upon which the detailed design will proceed.

12.6.1 Startup Accidents

"Classic" startup accidents resulting from a combination of operator error and failure of the startup instrumentation are considered here. The reactor startup procedure in the Mixed Spectrum Superheater consists of three principal stages executed in the following order (cf. Section 8.5):

1. Water is drained from the superheater.

2. The boiler is brought to critical and to full power.
3. The superheater is brought to critical and to full power.

A startup accident is possible during each stage. The accident during the first stage, i.e. during unflooding the fast core, is described in Section 12.6.7 under the general discussion of flooding effects. This accident is considered to be the "maximum credible accident" and its consequences are discussed in Section 12.7. The startup accidents for the second and third stages are described below.

The following accidents are selected for this conceptual design study to illustrate the principles upon which the final design will be based. In most cases, detailed studies of accidents have not yet been carried out. Confidence that the final design will meet the specified safety criteria is derived from preliminary studies and, where applicable, on the results of detailed studies conducted on similar BWR design.

Thermal Core Startup Accident

The startup accident in the thermal core is similar in nature to that in a conventional BWR for the following reasons:

1. The reactor will normally be started up by withdrawing control rods in the boiling section while the coupling control rods and fast core control assemblies are maintained in their least reactive positions. The thermal core during startup is, therefore, essentially separated from the fast core.
2. The fast core will be flooded prior to startup but will be drained before any attempt is made to move the boiler control rods. Thus, even though the fast core unflooding coefficient is positive, draining the fast core will not affect the thermal core startup accident. The flooding coefficient is expected to be negative so that accidental flooding during startup will tend to inhibit rather than enhance the startup accident.

A series of events has been hypothesized, which it is believed, represents the limiting class of accidents associated with starting the thermal core. The following operator errors and equipment failures are assumed to occur:

1. The range switches on the flux monitors are set on the power range scale instead of down-scale as required by operating procedures. Thus, short period scram protection is not available, and the reactor will not scram

until the power level reaches 120% of the rated power.

2. The withdrawal interlock circuits, which would prevent rod withdrawal unless at least two of the three flux level monitors are set down-scale, fail to function.
3. The operator fails to follow the pre-determined control rod patterns. He inadvertently picks the pattern resulting in an estimated maximum rod worth of 4% Δk .
4. The operator withdraws the maximum worth control rod at its maximum rate of 3 inches/second which adds reactivity at a rate of about 0.5% Δk per second.
5. The final shutdown results from a high neutron flux scram initiated at 120% of rated power. A delay of 0.2 second is assumed from the time of the scram signal to the time the rods begin to move.

It is assumed that the reactor is at a relatively low power level ($\sim 10^{-8}$ to 10^{-10} x rated) and at ambient temperature when the transient commences. These assumptions lead to a more severe thermal transient (higher peak fuel temperature) than would occur if it is assumed that the transient occurs at a higher moderator temperature. This results from the rapid formation of voids at high moderator temperatures tending to limit the rate at which reactivity can be added.

In this accident the reactor achieves such a short period that a very large overshoot occurs after the scram is initiated.

The turnaround mechanism is the Doppler reactivity effect. Formation of voids is neglected for the calculation.

Conclusions with regard to the accident postulated above are reached on the basis of similar transient studies conducted for the 50 MWe P.G. & E. Humboldt Bay reactor for which the maximum rod worth is 3.8% Δk and the maximum reactivity insertion rate is 0.43% Δk per second instead of the slightly higher values for the present MSSR design. The Humboldt Bay reactor analysis shows that the reactor period should reach a minimum in about 15 milliseconds. It is estimated that the maximum fuel temperature reached would be of the order of 1700°F at the hot spot which would not result in fuel melting or cladding damage. On this basis, it is expected that a similar startup accident for the MSSR prototype would not result in significantly higher temperatures than was calculated for Humboldt Bay and that fuel melting or cladding damage would not occur as a result of this accident.

It is pointed out that should later calculations show that the temperatures are higher than expected, sufficient flexibility is available for modification of the final design to control these temperatures. The primary variables available are the control strength and the maximum rate at which the rod can be withdrawn. The controlling physical criterion for the accident is that fuel cladding must not fail as a result of the accident.

Fast Core Startup Accident

The limiting hypothetical class of accidents associated with the third stage of startup, i.e. bringing the fast core to criticality and power, results from the following operator errors and equipment failures:

1. The reactor is started up in the normal manner by withdrawing control rods in the boiling section while the coupling control rods and fast core control fuel are maintained in their least reactive positions. The thermal core during startup is, therefore, essentially separated from the fast core.
2. The insertion interlock circuits in the fast core, which would prevent control assembly insertion unless a preset number of the flux monitors are set down-scale, fail to function.
3. Upon achieving criticality the operator continues to insert a single control assembly in the most important part of its stroke at its maximum rate.
4. The final shutdown results from a high temperature or high neutron flux scram initiated at 120% of rated power.
5. The superheater is at relatively low power (~3% of rated power) and low temperature when the transient commences.

This fast core startup accident has not yet been adequately analyzed. The control system philosophy will be to purposely limit the maximum insertion rate of the control assemblies such that in this hypothetical accident, the strong negative Doppler effect will be able to turn the transient around before final shutdown occurs.

A second fast core startup accident is also possible. If the fast core were started up inadvertently while the boiler was at very low power, the absence of steam would result in rapid heating of the fast core. This can have serious results causing meltdown of the fast core with the possibility of compaction and an increase in reactivity. To minimize possibility of such an occurrence, interlocks will be provided to prevent start-up unless the control

rods and elements are correctly positioned and to prevent movement of control elements from the superheater and buffer regions unless minimum steam flow is achieved. In addition, scrams will be initiated by temperature sensors at the outlet of the superheater core, and emergency spray cooling is provided to cool the fast core in the absence of steam.

12.6.2 Rod Withdrawal at Power

If the reactor is operating at rated power and, through error, the operator withdraws a control rod (or inserts a control fuel assembly into the fast core), and continues to move it, reactivity would be added and a power transient would result. As indicated above, if the control rod is in the thermal boiler, the negative void coefficient of reactivity would respond to the increased void formation to permit only a gradual increase in power rather than a fast period transient. Such a transient can easily be controlled by high flux scram instrumentation to prevent fuel cladding damage.

If the error occurs in the fast core, the limitation on the maximum rate of insertion will limit the power transient to rates that can be controlled by the high steam exit temperature and neutron flux scram instrumentation. Interlocks are provided in all cases to prevent withdrawal of poison or insertion of fuel with more than one rod at a time.

These accidents lead to less severe transients than for the startup accidents discussed above.

12.6.3 Cold Water Accident

Two accidents involving increasing reactivity by changing reactor water temperature are possible. These are:

1. If the feedwater valve were to be suddenly opened wide from an initially partially open position, the subcooling of water entering the core would undergo a moderate increase. This would cause an initial decrease in pressure from steam condensation, followed by an increase in pressure as subcooled water enters the core and increase reactivity. The transient is self-terminating at a higher than initial power level when the increased steaming balances the added reactivity from subcooling. The rate of reactivity insertion is proportional to the rate at which the feedwater valve is opened. For a given feedwater line size, the flux peak will increase as the time to open the valve decreases, all transients settling out at the same power. Increasing the line size tends to increase the peak flux and increases the final power. For cores of the design under consideration, the line size and opening rates are easily adjusted to

avoid significant fuel or cladding temperature transients.

2. A loss of electric load, resulting in actuation of the by-pass system, causes a loss of feedwater heating. The drop in feedwater temperature results in a reactivity-power transient. Although the temperature transient of the fuel has not been calculated, the transient heat storage capacity in the heater can be adjusted to provide a relatively large margin of protection from fuel cladding damage. In addition, by-pass lines are provided to shunt steam from the main turbine by-pass lines to the feedwater heaters. These smaller by-pass lines will help to maintain feedwater temperatures should the turbine by-pass valve be opened.

The criterion to be used in the final design for both cold water accidents is that no fuel cladding failure should occur even at the worst rate of reactivity addition.

12.6.4 Primary System Rupture

Double isolation valves in tandem are used to protect against primary system ruptures outside of the containment building. A rupture of the main steam line in the pipe tunnel would be the most severe accident of this type because ruptures in smaller lines would release less coolant from the reactor before isolation. Rupture of the main condenser diaphragm would cause closure of the turbine stop and by-pass valves due to loss of condenser vacuum.

If the main steam line should rupture outside of the containment building, the isolation valves would begin to throttle the flow significantly in about seven seconds and shut off completely in about ten seconds. The activity of the steam flashed from the reactor during this time would be predominantly N-16 and would be effectively dissipated by diffusion, decay, and expansion of the steam cloud before the cloud reached the site boundary.

If the steam line break should occur inside the containment building, the worst case would be if the rupture occurred at the point where the steam line leaves the pressure vessel. Approximate calculations for the blowdown rate show that the pressure will decay to 100 psig in about 25 seconds. The maximum pressure drop across the fast core during this blowdown period is estimated to be of the order of 750 psi. The fast core and its support structures are designed to withstand without collapsing pressures of this magnitude; however, some local permanent distortion can be expected. For example, the fast core shroud will collapse onto the fast core which would probably result in some difficulty in removing the fast core fuel. The primary concern is for the effect on reactivity of radial fuel

movements due to the tendency to compact the fast core. On the basis of present calculations, the inward movement of the fuel that could result from sudden decompression through the main steam nozzle appears to be small enough that the reactivity effects are not large.

Spray cooling is provided to remove heat from both boiler and superheating core if the normal heat removing capability should be lost. The primary source of the spray is the condensate storage tank with a capacity of 30,000 gallons. If this supply should be exhausted, the spray system is also connected to the fire protection system through a valve that can be turned on manually.

12.6.5 Loss of Primary Heat Sink

Loss of the primary heat sink could occur for three emergency conditions:

1. Closure of containment building isolation valves.
2. Closure of turbine admission valves accompanied by failure of the by-pass valves.
3. Failure of the main condenser.

In the first two cases, the emergency cooling system would be put into operation. This system and its operation is described in Section 4.0 under Reactor Auxiliary Systems. It is indicated there that the superheater fuel surface temperature rise will be less than 200°F during the transient period when the emergency cooling system is put into operation.

The third case, i.e. failure of the main condenser to remove heat might occur from loss of condenser cooling water. This loss would be detected by a gradual loss of condenser vacuum. A drop below a pre-set point will sound an alarm to alert the operator and a lower point will cause automatic closure of the reactor isolation valves and scram the reactor. This action isolates the reactor from its main heat sink (the condenser), and protects the condenser from being pressurized to the point of possible rupture. The reactor has an emergency condenser which is automatically placed in service as an emergency heat sink. Since this condenser is sized principally to remove decay heat, the reactor is scrammed, avoiding release of steam through safety valves. The scram signal may be initiated by a low condenser vacuum, simultaneous closure of turbine stop and by-pass valves, high reactor pressure or high neutron flux (high flux accompanies the rise in reactor pressure). The emergency condenser system is activated from closure of the isolation valves and/or high reactor pressure.

The criterion for the system design is to prevent the reactor from reaching a pressure-sufficient to lift the safety valves. In the limiting case, i.e., failure to scram on high flux or high pressure, the reactor would either finally be scrammed by low water level, manually inserting control rods, or by actuation of the Liquid Poison System.

The sequence of events would be similar to those just described if the condenser ruptured, except that some steam containing principally short lived N^{16} might escape into the turbine area before the stop and bypass valves closed. The hazard from radioactivity would be small compared to the thermal hazard and there would be little environmental hazard by the time the steam reached off-site areas.

12.6.6 Loss of Heat Sink with Fast Core Flooded

Normally, under shutdown conditions, the fast core is flooded so that no communication is available between the reactor and the main condenser. The decay heat during this time is removed by the shutdown cooling system. The shutdown cooling system, however, is a low-pressure system. If flooding occurs at high pressure or if the shutdown cooling system should fail, no heat sink would be available to remove decay heat.

To guard against this possibility, a small by-pass line with a meter-operated valve is provided to vent the the reactor steam dome to the emergency cooling system. The condensate from the emergency cooling system flows by gravity back to the reactor vessel so that water level in the vessel is maintained.

12.6.7 Flooding Effects

In earlier Mixed Spectrum Superheater designs, effort was made to make the reactivity effect of both flooding the superheater with water and unflooding the superheater as small as possible. It now appears that the effect of flooding the superheater can be made sufficiently negative that the likelihood that an accident can proceed in the reverse direction can be made very remote. This approach to the flooding problem is discussed below and is assumed for the accident analysis. Its advantage is that it is not sensitive to isotopic changes in the fuel which can occur after years of operation when high Pu-240 content fuel is recycled.

12.6.7.1 Large Negative Flooding Coefficient

It was previously believed desirable to follow the safety approach of thermal superheaters relative to the flooding problem. In thermal superheaters, the subcriticality in the flooded state is so small that inadvertent control motion

as a refueling accident leads to the possibility of criticality and power generation.

The situation in the fast superheater is quite different. Here a mechanism is available for introducing a large negative reactivity when the superheater is flooded.* On flooding a fast superheater, the neutron spectrum is degraded so that the absorption in the low epithermal energy region (meaning here from 1 ev to 10 kev) relative to the absorption in the high energy region is increased. Addition of parasitic absorbers whose cross section relative to the fissile materials increases in the epithermal energy region will result in a negative contribution to the reactivity upon flooding. By the addition of 3 weight percent europium oxide to the superheater fuel of the 75 MWe MSS, the reactivity change upon flooding the clean superheater is -8%. This same negative reactivity is achieved in the 500 MWe MSS by the addition of only .7 weight percent europium oxide. It is noted that addition of these strong epithermal poisons is permissible in a fast superheater since their effect during normal operation, i.e. in a fast spectrum is not great.

For fast superheaters using plutonium fuel, after the fuel has been irradiated for a relatively short time, an additional mechanism is created to introduce a large negative flooding effect. Plutonium-240 builds up in a fast reactor to about half the plutonium-239 concentration for the equilibrium plutonium cycle. On flooding the superheater, the strong Pu-240 resonance at 1.0 ev captures a sufficient number of neutrons which would otherwise have survived down to the highly reactive thermal region to introduce a large negative reactivity effect. A calculation of this effect for a 500 MWe MSS operating on the equilibrium plutonium cycle showed that the Pu-240 contributed a reactivity upon flooding of -7%. For a superheater loaded initially with U-235 fuel, the Pu-240 concentration in the equilibrium cycle is too low to contribute strongly to the flooding effect.

For both the U-235 and Pu-239 fuel cycles the buildup of fission products adds to the negative flooding coefficient. The absorption of the fission products relative to the fuel increases in the low epithermal energy region (from 1 ev to 10 kev) which is the region whose importance is most increased by flooding. Hence the fission products act as epithermal poisons.

* Reynolds, A. B., and Wolfe, B.: Flood Safety in Fast Spectrum Superheaters. Trans. Am. Nuc. Soc., 4, No. 1, p. 19, (June 1961)

By designing the superheater so that the reactivity change in flooding is large and negative, the fast core control system can be designed so that it is impossible to go critical in the flooded condition even if all the fuel control assemblies were fully inserted. The calculations referred to above indicate that negative reactivities of the order of -8% are achievable for fresh fuel by the addition of epithermal poisons and, in the equilibrium cycle, from Pu-240 and little or no added poison. The total excess reactivity in the unflooded reactor is 2.1%. Therefore in the flooded reactor the fast section would be 5.9 per cent subcritical even if all controls were in their most reactive positions.

12.6.7.2 Unflooding Accident

While draining the superheater, the shutdown control sheets are inserted, two fast core control assemblies are completely withdrawn, and two assemblies are "cocked" to control 1% reactivity, i.e. are nearly half inserted. Thus, after the core is completely unflooded and still at low temperature, the fast core is subcritical by at least 3.3% (cf. Table 3.5.3). During the unflooding process the core is always more subcritical than this. Such a large safety margin makes the achievement of criticality during unflooding extremely unlikely.

The water level will be lowered at a sufficiently slow rate that the maximum reactivity insertion rate is limited to a few cents per second. For example, if a ten minute period was required to drain the active region of the core, and assuming the reactivity effect of unflooding was +8%, the maximum reactivity insertion rate would be +5 cents/second. Even if gross errors in control positions occurred such that delayed criticality was achieved during the unflooding process, the positive period would scram the fast core control assemblies and coupling control rods.

Achievement of prompt criticality while draining the core, however implausible, is nevertheless credible. Theoretically a positive reactivity of 2.1% can be inserted into the fast core if the following errors or malfunctions occur simultaneously:

- a. The coupling control sheets are withdrawn,
- b. All four fast core control assemblies are completely inserted,
- c. All period scram and temperature scram devices fail to function.

The theoretical reactivity could be increased about 2.1% if the fast core had been mistakenly loaded with too reactive fuel, which could result from an enrichment error.

The consequences of this accident have not yet been analyzed, but they are probably not as serious as those considered in Section 12.7. (The calculations reported in Section 12.7 do not depend on the actual sequence of events of the maximum credible accident). The negative Doppler effect would compensate for the positive reactivity addition in the early stages of the accident to turn the initial excursion around since the autocatalytic effects due to boiling the water would act more slowly than the Doppler effect. As more reactivity is added (and still assuming that no rods scrambled) the reactor would eventually shut itself down by dispersion of the fuel. It is expected that detailed analysis of this excursion will show that shutdown occurs before the consequences considered in Section 12.7 are achieved.

12.6.8 Refueling Accident

The refueling accidents discussed here refer to accidents accompanying the loading or unloading of fuel assemblies.

The superheater is flooded for cooling and shielding during the refueling operation. The flooded core is subcritical by several percent even with all fuel control assemblies inserted (cf. Section 12.6.7). Reactivity could be added inadvertently during refueling in two ways. First, loading an assembly of higher enrichment than the design enrichment would add more reactivity than anticipated by the operator, although it is expected that even a very large enrichment error in a single assembly could not cause criticality. Second, reactivity could be added upon removing any of the first few fuel assemblies if, inadvertently, the thermal-neutron absorber had been omitted from the channel walls in the assemblies surrounding the removed assembly (cf. Section 3.5.9). Although the reactivity addition in this situation could be large, it would still require another simultaneous error (e.g. insertion of control assemblies) to result in criticality. To guard against this possibility, the size of the fast core fuel assemblies will be determined to limit the maximum reactivity that could be inserted upon removal of an assembly while flooded assuming that the thermal-neutron absorber has been omitted. Moreover, the resulting excursion, if it occurred, would probably shut itself off before complete unflooding occurred, because the unflooded core would lack the fuel assembly whose removal originated the excursion. Removal of the center assembly from the unflooded fast core would result in a reactivity loss of 1.5%.

In the thermal core more reactivity than expected may be added by loading an element with excess enrichment, by accidentally dropping an element into a region of high importance, or by removing poison. Normal refueling procedures exclude the first two, and all three conditions are guarded against by keeping a selected amount of "cocked" safety rods out of the core and coupled with in-core

neutron monitors. Thus the buildup of reactor flux above a predetermined level will cause the immediate insertion of additional poison into the core. The most serious accident, accidentally dropping an element into the most important region which can be assembled, can add reactivities up to $\sim 1.5\%$, not including the accompanying negative reactivity effects due to temperature rise and void formation, which would come into play while the element is falling if the controls failed to scram.

12.6.9 Fast Core Meltdown

Calculations show that an extreme case of melting in the fast core of the MSSR prototype could result in a reactivity addition of approximately $8\% \Delta k$. This calculation assumes that the entire fast core became molten and slumped so that all the steam void volume was instantaneously removed from the core.

The result of such compaction would be disastrous, but it is clearly hypothetical and would be impossible to achieve in practice. The heat generated by the mass as it approached critical would tend to disperse the fissionable material and stop the nuclear reaction well before anything approaching complete compaction could occur.

To obtain a more realistic estimate for the time scales involved in a core meltdown, a computer program was conducted to calculate the rate at which the fast core could become molten during a loss-of-coolant accident. In making the calculations, it was assumed that the reactor is at full power operating conditions. At some point, coolant is lost, and the system becomes adiabatic in the sense that no heat is removed from the fast core. Full power is maintained.

The results of the calculations (See Appendix, Section IV) show that the cladding would begin to melt at the hot spot in about three seconds after loss of cooling. The fuel meat would reach its melting point in about five seconds.

On the basis of these calculations, it appears that sufficient time is available for the control rods to scram and for the emergency spray system to be put into operation. It also appears that the reactivity input rates due to core compaction, if it compacts at the rate at which the core melts, may have serious consequences, but is not likely to result in a catastrophic explosion. This remains to be proven, however, and a computer program is being set up to attempt to calculate the nuclear reaction energy resulting from a core meltdown. In any case, it does not appear

likely that the energy release would be so violent, or so large in magnitude that it could not be contained by the vapor-tight containment vessel. This point is discussed further in the section on Maximum Credible Accidents.

12.6.10 Loss of Clad Accident

The loss of clad from the fast core fuel is related to the meltdown accident since melting of the clad and its consequent dropping out of the core might constitute the initial physical step in a meltdown excursion. The reactivity effect due to this loss of all the Hastelloy clad is -2.1%. This decrease in reactivity occurs because the negative reactivity effect of the increased leakage outweighs the positive effects of the hardened spectrum and reduced neutron absorption. The calculation was based on a twenty-group, one-dimensional model, including the effect of inelastic scattering by the clad. It was assumed that steam replaced the melted clad.

12.6.11 Fast Core Compaction

The potential reactivity effect of compacting the superheater was examined by considering that the fast core was pushed in radially displacing 3% of its volume (or about 17% of its steam volume). This is also equivalent to a reduction of the superheater effective diameter by 1/2 inch. No axial compaction was assumed. The entire superheater was considered to be compacted uniformly, with steam being displaced from the decreased volume.

The reactivity effect of a core compaction of this magnitude was calculated to be about +1%. This could be serious, but it should be noted that to achieve this reactivity input, the fast core must be uniformly compacted in the radial direction. It would be extremely difficult to do this, since the fuel channels will support without buckling pressures greater than the reactor operating pressure of 1,000 psig. Since the fuel rods themselves must be squeezed together to achieve a 3% reduction in core volume, the compacting forces must be considerably greater than 1,000 psi.

One can postulate an explosion in the thermal core due to a rod shoot-out, for example, which could exert compressive forces on the fast core. An accident of this nature would be initiated at a point in the thermal core, so that the forces exerted on the fast core would be applied unsymmetrically. This could tend to disperse rather than compact the fast core. Thus, it appears that even though the effects of core compaction on reactivity can be large, there is no plausible mechanism, short of core meltdown, by which a significant amount of compaction can occur.

12.6.12 Loss of Pressure in the Superheater

The criticality factor of the superheater is not strongly sensitive to variations in steam pressure. The reactivity change in the fast core due to complete loss of steam pressure was calculated to be only +0.05%.

Loss of steam is accompanied by an increase in leakage, which tends to reduce the reactivity, and hardening of the neutron spectrum, which tends to increase the reactivity. For the small size fast core of the 75 MWe MSS, the two effects almost exactly compensate each other.

Adequate cooling is maintained even though the boiler is shut down since the lower pressure would cause boiler water to flash into steam. In addition a drastic loss of pressure would automatically initiate spray cooling. It is possible that the small amount of water added by the spray cooling system adds a positive reactivity effect although early calculations* indicate that the maximum positive reactivity during a uniform flooding process is small.

12.6.13 Fuel Movement Due to Rod Vibration or Bowing

It would be possible to add reactivity to the fast core if the fuel was allowed to vibrate or bow. It is planned to prevent this fuel movement by the use of wire wrap to space the fuel rods. It is recognized that considerable experimental and analytical work will be required to give assurance that the fuel design is adequate. Provisions have been made in the Research and Development Program for an extensive experimental and analytical program to furnish guidance in this area.

12.7 Maximum Credible Accident

In order to evaluate the adequacy of the reactor containment, a highly improbable accident is postulated that would require a series of mechanical failures and a breakdown of the reactor protection system. The hypothetical accident is conceived without regard for the credibility of the sequence of events necessary to cause its occurrence. Its sole purpose is to facilitate the analysis to demonstrate that the public would be protected from exposure to radioactive materials for all reasonably conceivable occurrences.

* A. B. Reynolds, "Flood Safety of the Mixed Spectrum Superheater," GEAP-3737 GE-APED, P. 8 (May, 1961).

The nuclear energy released by the fast core excursion is calculated on a somewhat arbitrary basis to provide an order-of-magnitude estimate. The energy released is assumed to be equal to the energy required to heat the entire fast core to its melting point and to vaporize all the fast core fuel assemblies. These assumptions lead to an estimated nuclear energy release of about 8.9×10^6 Btu.

On the basis of work done on other fast reactor systems, there exists the possibility that a severe nuclear excursion in the fast core could result in an explosive release of energy equivalent to many pounds of high explosives. For the present reactor the credibility of an accident releasing significant explosive energy has not been assessed. Work is planned to analyse the magnitude of the explosive energy in the worst credible accident, and the biological shielding and missile shielding will be conservatively designed to withstand the calculated forces. The accident analysis for the all-fast reactor, EBR-II, shows that the explosive energy from several hundred pounds of TNT could be absorbed in this manner. It is therefore assumed that the biological shield and missile shield can be designed to protect the containment vessel from damage by missiles or shock waves.

In the present analysis, it is assumed that the excursion would be similar to those calculated for EBR-II. The analysis indicates that with extremely pessimistic assumptions, the explosive energy released in the worst credible accident may be equivalent to that from as much as 300 lbs. of TNT. The presence of the negative Doppler coefficient, the coupling to the thermal core, and the much reduced reactivity effect of compaction as compared to an all-fast reactor make this assumption conservative.

It is postulated that the explosive energy will rupture the reactor pressure vessel and release pressurized hot water and steam to the containment vessel. The accident is assumed to occur while the reactor is operating at a condition of maximum energy content. Significant coincident nuclear energy contribution from the boiler or chemical reaction energy is not considered.

The total enthalpy of the pressurized water and steam, corresponding to approximately 72,000 lbs. of saturated water, 2270 lbs. of saturated steam, and 310 lbs. of superheated steam, is estimated to be about 41.3×10^6 Btu. In addition, the sensible and decay heat released during the first two minutes in the boiler is of the order of 4×10^6 Btu. The total energy involved is therefore

Nuclear excursion	8.9×10^6
Thermodynamic energy	41.3×10^6
Sensible and decay heat	4.0×10^6
TOTAL	54.2×10^6 Btu

When this steam-water mixture is released to the containment vessel, along with the nuclear, sensible, and decay heat, the peak temperature and pressure in the building are estimated to be 250°F and 45 psia respectively. These figures are based on the free volume (80 per cent of total volume) of a cylindrical vessel 72 feet in diameter, 90 feet of straight length, and two hemispherical heads. The initial atmospheric state is assumed to be 14.7 psia, 100°F, and 100% relative humidity.

The peak pressures and temperatures calculated above will be developed within approximately one minute after pressure vessel rupture. After the initial peak, the pressure in the sphere will undergo changes with time as a function of two and possibly three competing mechanisms:

- a. There would be heat losses from the enclosure atmosphere to the enclosure shell, to the solid structure in the enclosure, and to the environment.
- b. There would be heat gain due to radioactive decay.
- c. There might be some heat gain from chemical reactions which might occur to a limited extent.

The net effect of these opposing mechanisms immediately after the accident would be to decrease the enclosure pressure. However, later radioactive decay heat would be generated faster than the enclosure could surrender heat to the environment, so that the pressure in the enclosure would tend to rise. Thus, a post-incident cooling system is provided to assure that the enclosure pressure would be reduced to about 3-5 psig by the end of the first day. Such pressure reduction reduces the amount of fission product material that might otherwise leak from the enclosure.

12.8 Radiological Effects

The radiological effects of the "maximum credible accident" are a sensitive function of the details of the plant design, the extent of credible fuel damage, and the meteorological conditions in the vicinity of the plant at the time of the incident. The objectives of final design and operating procedures will be to include measures to afford assurance that even in the event of the "maximum credible accident" no one outside the plant boundaries would receive more than the maximum permissible emergency dose of radiation.

12.9 Enclosure Design Criteria

Dry containment utilizing a vapor-tight enclosure has been selected for the Mixed Spectrum Superheater prototype. This was dictated by the experimental nature of the plant which will be used to demonstrate

the capabilities of the Mixed Spectrum concept. The experimental program will include, among other things, a series of tests to investigate the stability and response of the system. The prototype will therefore be subjected to operational conditions that would not normally be encountered in the operation of a power reactor. This factor, together with the unknowns that are present in the design by virtue of its novelty, were the primary considerations in specifying dry containment. Once the operating characteristics of the plant have become well understood, it is quite likely that the system will lend itself well to a pressure suppression type of containment.

Criteria to insure that enclosure integrity can be maintained in the event of a confinement requiring accident are similar to those used for conventional BWR's. They are briefly summarized as follows:

1. The enclosure design pressure (30 psig) is based on a hypothetical "maximum credible accident" which has an extremely low probability of occurrence.
2. The enclosure design incorporates the use of conventional safety margins and will be subjected to a series of pneumatic pressure tests up to 125% of design pressure.
3. The leakage rate for the enclosure is expected to be less than 0.5 per cent of its free volume per day. Actual value will depend on the site.
4. A containment spray system will be provided to help reduce enclosure pressure following a severe accident. The spray system will be operated manually and will include a tie-in to the fire protection water supply for emergency back-up.
5. All openings in the containment will be provided with means of preventing leakage if a severe accident should occur.
6. The concrete biological shield and missile shield above the reactor are expected to provide sufficient protection against missiles. If the accident occurs during refueling, the missile shield will not be in place. This situation requires further analysis, but on the basis of recent work carried out at the Stanford Research Institute, the probability of projecting missiles upward with sufficient to penetrate the containment appears remote.

12.10 Design Factors Affecting Safety

In general, the design of the plant will include features to assure that any release of radioactive materials to the environment during normal operation will be accomplished without exceeding the current recommendations of recognized national and international radiation protection groups.

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14.0 EXPANSION OF PLANT POWER OUTPUT TO 150 MWe

The present conceptual design for the MSSR prototype is based upon a power output of 75 MWe. It is planned to operate the plant at this power level to demonstrate the feasibility of the concept and to provide operating experience for this type of plant. Once the operating characteristics of the system are well understood, and provided that the operation has been satisfactory, it may be desirable to increase the power output. This will result in a net reduction of power costs, and at the same time brings the prototype closer to the operating conditions of the large plant.

In anticipation of future expansion, a number of provisions have been made in the present plant to facilitate the modifications necessary to increase the power to 150 MWe. The major change is the conversion of the prototype from a natural circulation to a forced circulation plant. This section presents some of the parameters for the 150 MWe core and describes the provisions made to expand the present plant capacity.

14.1 Parameters for 75 MWe, 150 MWe, and 300 MWe Plant

A comparison is made among the fast core parameters for the 75 MWe prototype, the 150 MWe expanded prototype, and the 300 MWe reference design (GEAP 3590)plants. The parameters are listed in the following table:

Parameter	75 MWe	150 MWe	300 MWe
Reactor nominal pressure, psi	1,000	1,000	1,500
Total Reactor Power, MWt	206	420	815
Fast Core Power, MWt	50	100	245
Fast Core Width, inches	32	32	42
Fast Core Length, inches	30	30	42
Steam Flow Rate, lbs/hr	570,000	1.14×10^6	2.8×10^6
Fuel Rod OD, inch	0.438	0.286	0.280
Pitch to Diameter ratio	1.05	1.13	1.14
Maximum Heat Flux, Btu/hr Ft ²	386,000	590,000	610,000
Hot Channel Avg. velocity, ft/sec	151	185	171
Hot Channel exit velocity, ft/sec	195	243	212
Max. surface temperature, °F	1,230	1,345	1,250
Max. fuel temperature, °F	4,320	4,680	4,500
Core pressure drop, psi	65	75	60
Power density, SpHR, KW/liter	98.5	197	217
Power density, BWR, KW/liter	20.4	35.5	35.7
Steam exit temperature, °F	950	950	950

14.2 Provisions for Expansion

To facilitate conversion from 75 MWe to 150 MWe operation, the following features have been provided in the present plant design:

1. Forced Convection Recirculation Loops: Four nozzles have been provided in the pressure vessel for the addition of two recirculation loops. Pipes attached to the nozzle will be capped off on the outside face of the biological shield. Space is provided in the reactor building for the recirculation pumps and associated equipment.
2. Primary Coolant System Piping: The feedwater lines and main steam line have been sized to take the higher flow rates associated with 150 MWe operation.
3. Turbine-condenser Plant: It is planned to add a second 75 MWe turbine-generator system for the higher power. Space has been allowed on the reactor site adjacent to the present turbine building.
4. Nuclear Instrumentation: Consoles, channels, monitoring stations, etc., have been designed for expansion to 150 MWe operation.
5. Auxiliary Equipment: It is planned to double capacity by paralleling all existing auxiliary equipment such as clean-up systems, heat exchangers, dump tanks, etc. Space is available in the reactor building to do this.
6. Boiling Core Layout: Space has been provided in the thermal core to increase the number of fuel assemblies from 128 to 156. The locations for the 28 additional fuel assemblies will be occupied by dummy assemblies for 75 MWe operation.

It is recognized that the provision of these features would result in a higher cost for both 75 MWe and 150 MWe operation than would be the case if the plant were specifically designed for a single rating only. At the present stage of development, however, it appears to be advantageous to have available the additional flexibility.

14.3 Other Modifications Required

In addition to the provisions incorporated into the plant design, other modifications would be required to convert the plant to 150 MWe operation. The primary changes, in addition to those discussed above, would be the provision of internal steam separators and a redesign of the fast core fuel.

The addition of radial-type steam separators is not expected to be difficult, since space is available above the present core. The separators would replace the chimneys in the present design and would be supported by the flange upon which the chimneys rest.

The redesign of the fast core would involve a change in the fuel rod diameters and rod spacings as shown in the table of Section 14.1. The smaller rod size would result in higher fabrication costs, but this will be offset somewhat by the larger pitch-to-diameter ratios. The core parameters listed in the table for 150 MWe assume that the fuel can be operated with a surface temperature of 1350°F.

On the basis of the brief study undertaken to investigate the possibilities of higher power operation, it appears that sufficient margin is available in the present pressure vessel to double the power output. As pointed out above, one of the critical items will be whether the fuel cladding can be operated at 1350°F.

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15.0 RESEARCH AND DEVELOPMENT PROGRAM

15.1 Introduction

The design of the prototype discussed in this report involves a number of technical areas in which there is little experimental data. In these areas design assumptions have been made which require experimental verification.

This section outlines the Research and Development Program aimed at providing the data necessary to clear up the major problem areas presently recognized.

A tentative time schedule leading to the construction and operation of the MSS prototype is also included. It should be emphasized that the schedule shown is a minimum schedule in the sense that it assumes successful results for the development program and no major delays as a result of false starts. It also assumes that problems, not now apparent, which are uncovered as the work progresses can be solved concurrently with the planned development, design, and construction effort.

15.2 Fuel Element Development

The objective of this task is to develop a suitable fuel element for the superheating portion of the Mixed Spectrum Superheater. The fuel element must be able to withstand long burnups and heat transfer to high-velocity superheated steam with satisfactory mechanical stability. The fuel development program is divided into the following major sub-tasks:

15.2.1 Cladding and Structural Materials

At the present time it is not clear what material will be most suitable for use as cladding for nuclear superheater fuel. The purpose of this task will be to investigate various cladding materials from the standpoints of, for example, stress corrosion, erosion, compatibility with fuel, strength, ease of fabrication, nuclear properties, and fuel cycle economics. Materials of particular interest are Hastelloy, Incoloy, and other high-nickel alloys, although other materials are also being considered. Materials for structural use other than for cladding will also be investigated. In particular, information regarding the feasibility of incorporating thermal or epithermal poisons in fuel channel materials is desired.

The work under this task will include a continuing survey of developments in the materials field to take advantage of the work done at other installations. It appears, however, that only limited data are available for conditions which duplicate the environment to be expected in nuclear superheat plants and extensive experimental work is required to obtain such information. Examples of the type of work required are some specific experiments presently being planned, which include irradiation of a rod cluster simulating the Mixed

TABLE 15.1

MIXED SPECTRUM SUPERHEATING REACTOR DEVELOPMENT SCHEDULE

	FY 1962	FY 1963	FY 1964	FY 1965	FY 1966	FY 1967	FY 1968
75 MWe Prototype Conceptual Design	_____						
Critical Experiment Detailed Design		_____					
Critical Experiment Construction		_____					
Critical Experiment Operation			_____				
General MSSR Design and Economic Studies	_____				_____		
75 MWe Prototype Scoping Design		_____	_____				
75 MWe Prototype Detailed Design			_____	_____	_____	_____	
75 MWe Prototype Construction				_____	_____	_____	
75 MWe Prototype Startup Operation						_____	_____
500 MWe Power Plant Development Studies					_____	_____	_____

Spectrum fuel in the E-SADE loop and short term high heat flux experiments in the GETR Trail Cable facility to study UO_2 -clad interactions. Experimental data must be obtained to determine the stress-corrosion resistance and thermal cycling behavior of high-nickel alloys at high temperatures under superheated steam conditions. Ability to retain strength and ductility under irradiation is not known for the large integrated fast flux doses to be expected. Means of obtaining this information will be considered, possibly by irradiating samples in existing fast reactor facilities.

15.2.2 Prototype Fuel Design

Under this sub-task, the mechanical design of the prototype fuel will be carried out. The various parameters affecting fuel performance, such as plenum voids, UO_2 -clad interactions, clad strength, spacers, fission gas release, etc., will be analyzed to arrive at an optimum design. Once the design has been established, effort will be applied toward the design and analysis of in-pile irradiation test specimens as final proof tests before reactor operation. The necessary heat transfer, fluid flow, and safeguards analysis for the irradiation tests will be carried out under this sub-task.

15.2.3 Fuel Fabrication

Fabrication methods for manufacturing fuel elements for the superheating portion of the MSSR must be developed. Various mechanical configurations will be analyzed and tested to determine the most practicable means of providing accurate spacing between fuel rods and for assembling the rods into bundles. The heat transfer, flow, and vibration tests described elsewhere are expected to provide guidance for this task.

The feasibility of incorporating thermal and epithermal poisons into the fuel elements and fuel channels must be firmly established. Some experience is available, but knowledge must be extended to meet present needs which are more stringent than have been the requirements in the past. Irradiation behavior of the fuel with poisons must be determined. This can be accomplished both in short irradiation screening tests and long term capsule test. This will require in-pile testing as discussed under sub-task 15.2.4 below.

A part of the work under this task will include coordination with other programs (such as the Fast Ceramic Reactor program) to evaluate the irradiation behavior of the proposed superheater fuel. Also, it is felt that the fabrication problems encountered in handling plutonium can best be investigated in conjunction with other programs.

15.2.4 Irradiation Tests

The irradiation tests will be divided into two portions. The first part will involve irradiation of single rods or small clusters of rods and small UO_2 -clad samples. Tests on fuel rods clad with

Hastelloy and Incoloy are planned as a first step in this direction. The purpose of these tests is to obtain preliminary information regarding possible fuel-clad interactions and the behavior of these materials in a nuclear superheated steam environment. If no inherent weaknesses are revealed by these tests, and the results of the other parts of the fuel study continue to show that Hastelloy or Incoloy are compatible with use as superheat fuel cladding, effort will be concentrated on the development of these materials. If the results are not favorable, additional materials will be studied and tested.

The second part of the irradiation tests will be in the nature of proof tests of the prototype fuel. They will therefore come later in the development program after the materials and fuel design have been determined. Burnup data and irradiation behavior, particularly with regard to phenomena such as fuel redistribution and epithermal poison migration will be obtained.

One problem with regard to irradiation proof testing of prototype fuel arises because the superheat fuel must operate in a fast neutron spectrum for which no test facilities exist at present. This problem has not been satisfactorily resolved as yet, and accurate simulation of MSSR fuel under environmental conditions may be difficult. Further studies are required. It is possible that the present development program may reveal that total reliance must be placed on analytical studies and irradiation tests run in thermal reactors. The prototype MSSR will then provide the bulk of the burnup and irradiation fuel testing.

15.2.5 Heat Transfer Tests

Under this task, tests will be performed to see the effects of close packing and methods of fuel rod supports on the heat-transfer properties of the fast core. Of particular interest are the peripheral variations in surface temperature for individual rods as a function of rod spacing. This is a relatively unknown area at present, particularly for the close rod spacings under consideration. Experiments are being conducted at other installations, notably at General Atomic and Aerojet-General, and this task will include coordination with the efforts of these workers and others in this field.

The proposed tests represent an experimental study of heat transfer to superheated steam using a bundle of rods in a hexagonal arrangement. Combinations of geometrical parameters (rod diameters and pitch-to-diameter ratio) will be tested. The rods will be heated electrically. Test runs will be made at various flow rates and heat fluxes. Steam velocities and heat fluxes characteristic of reactor operation will be attained.

Each rod bundle will be instrumented to measure rod surface temperatures as a function of both circumferential and axial position.

15.2.6 Flow and Vibration Tests

This task will attempt to explain quantitatively the parameters that influence the vibration of multi-rod assemblies in parallel flow. Measurements of vibrations of 0.280 inch O.D. and 0.500 O.D. fuel rods will be made in superheated steam with full-length, multi-rod assemblies. Maximum amplitude, frequencies, and modes of vibration will be detected for velocities of 50-200 fps, 1.0 to 2.5 lbs/ft³, and several wire-wrapped spacer pitches. The data will be analyzed to relate results to various reactor operating conditions and possible design revisions. An attempt will be made to analyze pressure drop fluctuations as a function of instantaneous displacement. High-frequency-response pressure difference measurements across and along the fuel element will be measured. Average pressure drop measurements will be obtained also.

Tests similar to those mentioned above are presently in progress at APED under the base superheat development program. Pressure drop measurements and frequency response information, in particular, are being developed in connection with the fretting corrosion program. It is planned to utilize the results of these and other applicable developments to assist in carrying out the present program.

15.3 Emergency and Shutdown Spray Cooling

Spray cooling is being considered for emergency measures in the event of loss of steam in the superheating portion of the Mixed Spectrum Superheater. The purpose of this task is to study the effectiveness of spray cooling to establish the adequacy of such a scheme.

To simulate the fast core geometry, a hexagonal array of at least 19 rods, 30 to 40 inches in heated length, is proposed. A similar but shorter bundle of unheated rods will be placed above the heated rods to simulate the upper reflectors. Spacing will be accomplished by wire-wrapping two out of every three rods. It is planned to run tests at 1,000 psia for several different spray nozzles at various flow rates, degrees of sub-cooling of spray water, and heat fluxes.

The rods will be instrumented to measure temperatures on the inside surface of the heated section. The spray distribution will be made with the system cold. The heat transfer regime will be deduced from temperature readings with the system hot. It is planned to analyze the test data both to establish the adequacy of the spray cooling system and to attempt to establish a basis for predicting the behavior of similar systems.

15.4 Mechanical Development

There are at least four major areas of mechanical design of the Mixed Spectrum Superheater that will require development work. These areas are: design of the fast core shroud and support structure, control element guidance in the fast core, control rod drive design for the fast core and thermal buffer, and superheater outlet seal. These problems are subdivided into the following tasks:

15.4.1 Fast Core Shroud and Support Structure

In the present conceptual design, the fast core is supported by four columns that form the corners of the superheating section. The weight of the core and the forces due to pressure drop are transmitted to the vessel bottom through these four corner posts. The fast core shroud consists of thin plates between the corner columns as described in Section 3.1.3. The plate will deflect under the differential pressure that exists during operation, so that the forces are carried by the fast core fuel assemblies tending to hold them firmly in place. This task is aimed at experimental verification of this design concept. The problems foreseen are:

- a. Achieving a symmetrical deformation so as to not upset fast core alignment.
- b. Keeping stresses low to insure high reliability and prevent buckling of the plates and columns.
- c. Determine the effects of thermal stresses and thermal and mechanical cycling.

The development program planned to attack these problems is divided into the following sub-tasks:

- a. Analyze the present design to assure that best design is used.
- b. Design experiments or simple tests where necessary to provide information to carry out (a) above.
- c. When design is finalized, design a mock-up to check analytical procedures.
- d. Analyze the results to mock-up tests to check adequacy of design and to provide a basis for design revisions.

15.4.2 Fast Core Control Element Guidance

Although the control elements will operate inside guide tubes which are sufficiently strong to resist with minor deformation the clamping action of the differential pressure across the flexible wall, there is a possibility that the clamping action could cause some misalignment of the guide tubes. Misalignment caused by thermal gradients could also contribute to this problem. The purpose of this sub-task is to provide assurance that the control elements will move freely within the guide tube under all anticipated conditions. The following problems will be considered.

- a. Analyze effect of fast core shroud deflections on control fuel assembly movements.
- b. Analyze effect of thermal gradients on control fuel assembly clearance to see whether movements are restricted.

- c. Analyze effect of possible misalignment of control fuel bundles with their respective drive shafts as a result of fast core shroud deflections.
- d. Design coupling between control fuel assemblies and drive shafts to eliminate effect of misalignment.
- e. Design a roller or other guiding mechanism to assure that control fuel assemblies will move freely within their guide tube under all operating conditions.
- f. Build a prototype or mock-up to proof-test selected design.

15.4.3 Fast Core and Thermal Buffer Control Drive Design

The control elements in the fast core and thermal buffer are activated by control drives mounted above the reactor top head. The present MSSR conceptual design specifies the use of twin-screw type of drive mechanism similar to those used in existing small BWR's, for example, JPDR and RWE. These drive designs must be reviewed to determine whether they will be satisfactory for this application. It is expected that some development will be necessary, particularly with regard to the seals where the drives penetrate the vessel. In addition, the means of coupling the control element to the drive shaft to allow for possible misalignments as discussed under Section 15.4.2 must be developed. The breakdown of this task is as follows:

- a. Review the existing twin-screw drive design and modify where necessary.
- b. Review other drive design, for example, the magnetic jack drive, for possible application to MSSR.
- c. Test the resulting design.
- d. Evaluate results and write design report.

15.4.4 Superheater Outlet Seal Development

The design of the seal arrangement to accommodate differential thermal expansions in the superheater outlet pipe is described in Section 3.1.4. The use of sliding seals for this application (i.e., large diameters, relatively low leakage, and high-temperature steam environment) is not too common, but large-diameter dynamic seals are presently in use in many applications, for example, as piston ring seals in large diesel engines and shafts seals for steam turbines. It is, therefore, felt that commercially available seals can be adapted to the present design, and a large development effort does not appear to be necessary.

Reliability of the seal will have an important influence on the economics of the MSSR, since the location of the seal precludes maintenance while the reactor is in operation. The seal must therefore function for long periods of time with little maintenance and must be designed so that small leaks will not require shutdown of the reactor.

To provide these features, existing seal arrangements will be studied to determine the most suitable design. Once the design has been selected, tests will be run either on a mock-up or a prototype to check its operation. In addition, the entire seal arrangement consisting of seals, piping, and flanges must be designed so that it can be assembled and disassembled during a normal reactor shutdown period, and the components must be readily available for servicing.

15.4.5 Other Mechanical Design Development

In making the conceptual design study of the 75 MWe MSSR prototype, it was apparent that the sub-tasks described above are the critical ones as far as development is concerned. As the detailed design is undertaken, however, it is expected that a number of less obvious problems will be brought to light. In this category are such items as valves for special application, refueling mechanisms and procedures, provision of instrumentation for experiments, etc. These and other mechanical design studies will be conducted as further work shows the necessity of insuring performance of mechanical components.

15.5 Reactor Stability Studies

Although conventional boiling water reactor systems have exhibited adequately stable and non-oscillatory behavior over the required range of operating conditions, the performance characteristics of the proposed Mixed Spectrum Reactor should be studied to provide assurance that the reactor will operate in a stable manner. BWR stability studies currently in progress at APED and elsewhere will be utilized to provide a basis for planning the work for this task. The understanding of BWR stability phenomena derived from these studies is expected to minimize the effort necessary to assure that the MSSR stability characteristics are basically predictable. The initial effort under this task will therefore undertake to apply available data to the Mixed Spectrum concept.

On the basis of these initial studies, an analytical and experimental program will be prepared for the prototype reactor. The objective will be to identify, analyze, and evaluate the problems in Mixed Spectrum reactor dynamics to assure safe operation of the proposed plant.

15.6 Reactor Safety

The Mixed Spectrum Superheater combines a fast and thermal reactor in the same pressure vessel. The analysis to date indicates that accidents common to both boiling reactors and those common to fast reactors must be considered. The accidents involving the fast superheating section are of primary concern.

Preliminary hazards considerations indicate that the plant can be designed so as to satisfy normal safeguards requirements. However, based on the experience of previous fast reactors, it is expected that considerable detailed analysis and some experimental data will be required in order to provide confidence and satisfy license requirements. This task is aimed at meeting this need for data.

As the first step, a computer program is being written to examine in detail the meltdown accident. Further work on the analysis of other mechanisms which can lead to compaction is also planned. The detailed analysis of a startup accident in which all control rods are removed and the superheater section is unflooded will be assessed. The results of the heat transfer and fluid flow work, and in particular the results of the spray cooling experimental work will be quantitatively assessed. Modifications to the design will be suggested in order to assure that even under extreme conditions the integrity of the reactor is preserved.

Activities in this area are expected to continue throughout the development and design program and up to the issuance of the final operating license.

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16.0 MSSR CRITICAL EXPERIMENT

Various reactivity and criticality parameters must be measured experimentally to provide the degree of confidence required to proceed with the detailed design and construction of the prototype reactor. As noted in Section 3.5, the physics of the MSS is complex. Since the feasibility and economic incentive of the MSS concept depend upon the physics analysis, a critical experiment is planned to confirm the calculational methods used and to measure the important parameters in a system sized similar to the prototype. The experiments planned include measurements of the Doppler coefficient, flooding and unflooding coefficients, power shapes, effectiveness of each set of control elements, critical masses for various loadings, and the degree of nuclear coupling between the fast and thermal cores.

This section describes the conceptual design of the critical experiment. The mechanical design, loading procedures, and preliminary safeguards considerations are discussed.

16.1 Description of the Facility

The Mixed Spectrum Superheater Critical Experiment (MSSCE) is a zero-power mock-up of the core of the 75 MWe MSS prototype. It is, however, not a duplicate and, in particular, to reduce costs, it is planned to utilize available AEC critical fuel for the thermal buffer and boiler. As in the MSS, the critical experiment core will consist of four annular regions: an inner unmoderated superheating region, an inner unmoderated buffer region, an outer moderated buffer region, and an outer moderated boiling region similar to a conventional BWR core mock-up.

An elevation view of the facility is shown in Figure 16.1.1. A cross-sectional view of the core is shown in Figure 16.1.2. The outer moderated regions consisting of the thermal buffer and thermal core are contained in an 11-foot-OD by 8-foot high aluminum tank raised approximately 6.5 feet above the floor. The tank has a 33-inch square hole down its center in which the unmoderated core is located. The hole has a double wall which is provided to assure that the fast core will not be flooded if the outer wall should rupture.

The thermal buffer consists of slightly enriched uranium rods clad with stainless steel and arranged two fuel cells deep around the center hole. The thermal core is made up of 2% enriched UO_2 fuel, approximately 6 feet long. As noted, this fuel will consist of elements supplied by the AEC from other installations. For shutdown or scram, twelve $1/8$ inch thick boral sheets are gravity-dropped into the moderated core. Eight of the sheets are arranged uniformly throughout the thermal core, and four are in the thermal buffer adjacent to the fast core (one on each side of the square center hole). Control of the thermal core will be primarily by varying the water level. Scram action of the boral

sheets is accompanied by simultaneous dumping of the water to reduce reactivity. Water flow into the tank is orificed to limit the maximum rate of reactivity addition.

The fast buffer consists of 32 (four of the corner assemblies are made larger to include the corners) fuel assemblies mounted on the inside of the double wall between the fast and thermal cores. The buffer fuel assemblies are suspended from brackets on the wall and are locked in place with cams or wing nuts. The entire fast core must be removed before the fast buffer can be moved. This is to prevent accidental removal of the buffer, which could result in a significant shift in power shape. The fast buffer fuel assemblies, which will be loaded manually prior to loading the fast core, consist of 26 natural uranium (UO_2) rods clad with 0.450 inch OD stainless steel tubing. The cladding wall thickness is 0.030 inch. Each fuel assembly is contained in a stainless steel channel approximately 3-1/2 inches by 2 inches in cross section.

The inner unmoderated region corresponding to the fast core of the MSSR is approximately 29 inches square in cross section. It is made up of 68 fuel assemblies, each of which is 3.58 inches square and 54 inches long. Each assembly is made up of 52 enriched uranium (20-25 per cent) oxide fuel rods 0.450 inch OD with 0.030 inch stainless steel cladding. The active length of fuel is 30 inches with a 12-inch reflector of depleted uranium or stainless steel on each end. Control of the fast core is achieved by attaching two of the fuel assemblies at the top to drive screws. Scram or shutdown is achieved by accelerating downward four additional fuel assemblies by the use of pneumatic drives at the top of the core. The fuel assemblies used for control and scram are somewhat smaller than the regular fuel assemblies to allow space for guide tubes. Additional shutdown margin is provided by the four coupling control sheets in the thermal buffer.

To protect the reactor from a meltdown during an uncontrolled excursion, a number of the fast core fuel assemblies are provided with U-235 metal fuses. The fuses are shown in the sketch in Figure 16.1.3. In an excursion, the fuse would heat up faster than the fuel so that it would melt and allow the fuel assembly to fall out of the fast core. A concrete cone situated on the floor below the fast core assures that the fuel assemblies would be dispersed upon falling. It is planned to load the fast core fuel assemblies remotely by the use of a motorized torque and speed-limited winch and cable arrangement. The loading operation will be viewed remotely with a closed circuit TV camera.

16.2 MSSCE Preliminary Hazards Analysis

It is planned to use the existing Critical Experiment Facility at the Vallecitos Atomic Laboratory of APED to perform the Mixed

Spectrum Superheater Critical Experiment. The following discussion is a preliminary hazards analysis intended to demonstrate that the facility can be used safely for this purpose.

16.2.1 75 MWe Prototype Fast Core Parameters

The important physics parameters for the fast core of the 75 MWe prototype MSSR that will influence the design of the critical experiment are listed in Table 16.2.1. The dimensions of the critical experiment are somewhat larger than those listed for the prototype to allow for the difference in fuel densities. The prototype fuel is expected to be UO_2 of 95.5 per cent theoretical density, while the critical may utilize densities of about 90 per cent of theoretical. The dimensions of the critical are enlarged to keep the total weight of fuel approximately the same in both cases.

16.2.2 Comparison Between MSSCE And Previous Critical Experiments

To provide a basis for evaluating the hazards associated with operating the Mixed Spectrum Critical Experiment, it is useful to compare the features of MSSCE with those of the existing BWR critical which has operated successfully for a number of years. It would thus be possible to direct attention primarily to the differences between the two facilities that would tend to contribute to the hazards of the MSSCE.

The differences between the MSSCE and the previously operated BWR criticals stem from the presence of the fast section in the center of the core. This region is not cooled, and it has a smaller Doppler coefficient and shorter neutron lifetimes than the BWR cores. In addition, because it is a fast core, the plausibility of a catastrophic explosion accident of the type possible in a small metal assembly must be investigated in detail. This point is discussed in the next section.

In evaluating the maximum credible accident for the BWR criticals, the ultimate shutdown was always boiling. The clad integrity was maintained, and the reactor, if not completely shut down after the expansion, could operate indefinitely in the boiling condition where it would be self-controlled and at a relatively low power.

For the MSSCE core, there is no moderator in the fast region; hence the ultimate shutdown is disassembly of the core. To insure rapid disassembly of the core, certain elements are fused so that they will fall out of the core in an excursion before gross fuel damage occurs.

TABLE 16.2.1

75 MWe Prototype Fast Core Physics Parameters

Fast Core Dimensions (64 bundles)

Active fuel length, inches	30
Cross-section, inches	27 3/4" square

Fast Buffer Dimensions

Active Fuel length, inches	30
(12 inches of depleted uranium reflectors are placed above and below fast core and buffer)	

Volume fractions (fast core, buffer, and reflectors)

UO ₂ fuel	0.485
Stainless steel clad	0.165
Structural materials	0.178
Steam	0.172

Fuel Enrichments

Fast core	23% enriched
Fast buffer	1.5% enriched
Reflector	0.3% enriched

Reactivities - Fast Core Only

Doppler at room temperature with "steam" present	$-10^{-5} \Delta k \text{ per } ^\circ\text{C}$
Removal of most important central fuel assembly of 64 in fast core	minus 1.3% Δk
Removal of least important fuel assembly	minus 0.3% Δk
Compact the fast core (uniformly reduce the effective radius 1 inch, height of core stays the same)	plus 4.7% Δk

Meltdown

Reduce height of core by 17.2% (i.e., remove all steam, leave radius same). Top reflector stays with top	$\Delta k = +8.0\%$
Loss of clad	$\Delta k = -2.0\%$
Loss of steam	$\Delta k = -0.4\%$

Flooding effect for 23% void (steam volume) is positive

Although the MSSCE does have a light-water-moderated region, the reactivity effects in that region have only a small effect on the overall reactivity of the core. It follows that if excess reactivity is added to the thermal region, the reactor would behave as a water-moderated reactor. If, however, excess reactivity is added to the unmoderated region, the thermal region cannot be relied upon to shut down the reactor. The following discussion deals only with reactivity additions to the fast region.

16.2.3 Excursion Types

Nuclear excursions are caused by reactivity addition. The discussion of maximum credible accidents falls naturally into two parts. The first part (Section 16.2.4) covers the effects of reactivity additions, and the second part (Section 16.2.5) covers the credibility of high rates of reactivity additions.

The effect of reactivity addition depends upon the initial power level of the reactor. At very low initial power levels the statistical assembly accident becomes possible. This refers to an accident which can occur with a small metal assembly which has a very small inherent source of neutrons. In such a reactor a large amount of reactivity can be added during the time in which no neutrons happen to be present. The source of neutrons in a small fast assembly due to spontaneous fission of U-238 and U-235 is about 10^{-5} of the source present in the Mixed Spectrum Superheater critical core. Therefore the time required to establish a persistent chain reaction in the Mixed Spectrum Superheater critical core will be about 1/1000 that of a small fast assembly. For small fast assemblies the time between reactivity addition and beginning of neutron multiplication has been observed to be as long as a few seconds. This means that the statistical lag between reactivity addition and the following of this by the neutron flux will be of the order of a few milliseconds in the Mixed Spectrum Superheater Core. Thus the statistical assembly accident is not of major importance in the Mixed Spectrum Superheater core, since credible reactivity addition mechanisms add negligible reactivity in this short period.

Accidents will be considered which are caused by ramp additions of reactivity. The inherent source of neutrons is small. Hence the reactor will remain at a very low power level for some time after the initiation of the ramp. By the time the power reaches a level high enough to heat the fuel the reactivity may exceed prompt critical. The maximum power of the resulting excursion is limited by the prompt negative fuel temperature coefficient, i.e., the Doppler effect.

Accidents which are caused by a very rapid ramp, i.e., a step increase in reactivity, can be tolerated only if the amount of reactivity that can be inserted is limited. For the Mixed Spectrum Superheater this amount would be specified as a limitation on experimental apparatus such that only \$1.00 can be credibly inserted rapidly.

16.2.4 Calculated Excursions

The maximum temperature of the fuel is a function of the ramp rate and the Doppler coefficient. Three ramp rates have been investigated to determine the history of the power during excursions. For these calculations the Doppler coefficient at room temperature is assumed to be $10 \times 10^{-6} \Delta k/^\circ\text{C}$ and a $1/T$ dependence of Doppler coefficient is assumed.

The only shutdown mechanism assumed in the calculation is the Doppler effect.* The three ramp rates, which were investigated, are .05, 1, and 10 \$ per second. The table below summarizes the results of the problems.

Ramp Rate	Time to 500°F	Temperature 0.03 second after 500°F	Temperature 0.1 second after 500°F
.05 \$/sec	20.5 sec	530°F	600°F
1 \$/sec	1.2 sec	1200°F	1500°F
10 \$/sec	.16 sec	12,000°F	16,000°F

The 10 \$/sec accident is included for completeness, but is not considered credible.

In the first two reactivity ramps considered, the power rises relatively slowly until prompt critical is reached. Beyond this point, the power rises very rapidly so that the fuel attains the temperature shown in the table in the times indicated.

The excursion is turned around by the negative Doppler coefficient, and the power drops off rapidly. Following the initial pulse, the power would again rise slowly for several seconds. By this time, however, the fuses will have melted and made the core subcritical by dropping fuel assemblies out of the core.

* Actually, the reactor would shut down by disassembly of the core due to melting of a U-235 metal pin in certain fuel bundles. Part of the core will fall, thereby decreasing reactivity. After melting the fuel pin, approximately 0.1 second is required for 2 inches of motion, at which point the reactivity decrease becomes significant.

It can be seen from the above table that there is not a great difference between the temperature at .03 second and at .1 second, so that spring loading to obtain an acceleration of 10 g would not significantly increase the safety of the reactor.

The vaporization temperature is the upper limit to the safe temperature of the fuel. Above that temperature an explosion is possible. For UO_2 the vaporization temperature is approximately 5700°F. Hence, it is estimated that, for a Doppler coefficient of $10 \times 10^{-6} \Delta k/^\circ\text{C}$, a ramp of approximately 6 \$/sec would not cause an explosion. It is expected that, in the AEC License Application Amendment, the maximum credible accident would have a ramp rate of 0.5 \$/sec, and the required minimum negative calculated value of the Doppler coefficient at room temperature would be $5 \times 10^{-6} \Delta k/^\circ\text{C}$.

16.2.5 Maximum Credible Accident

The credibility of rapid insertion of reactivity depends on consideration of the mechanisms of reactivity addition in light of the proposed design.

Large rates of reactivity addition require large velocities. These velocities can conceivably arise from falling objects. Heavy equipment falling on the core from heights greater than about 10 ft. could cause rapid compression for a core supported from below; however, the Mixed Spectrum Superheater core has no bottom support. The fuel will be hung from an upper grid sheet. The possibility of the entire core falling intact on a flat surface and compressing is not considered to be credible because of concrete cone under the core.

Objects falling on top of the core do not add reactivity by improved neutron reflection because the fast core already has a thick U-238 (or stainless steel) reflector on top and bottom.

Fuel elements cannot be inserted from above through the upper support plate, so that falling fuel cannot enter the core intact. Fuel elements will be loaded from below and inserted with a mechanical lift due to their heavy weight (about 175 lbs.). Maximum ramp is limited by the mechanical limitation of the lift.

With the fast core designed so that the fuel upon melting will fall out of the core, an accident in which a small excursion melts the core which then flows into a supercritical configuration is not considered credible. In addition, a number of the fuel

assemblies will have built-in metallic U-235 fuels which will drop fuel out before the fuel itself can melt.

As shown in Figure 16.1.3, the top end of each fuel assembly will be designed so that water leading into the core from above will flow between the fuel assemblies onto the floor. Inadvertent flooding of the fast core therefore is highly unlikely.

The assumption is made that a core can be designed which cannot be jammed. Clearances will be large enough to allow for all plausible warpages and distortion. Several tentative designs have been suggested which appear to satisfy the requirements. To prevent fuel from being jammed into the core during loading, the loading machine will be driven by a torque-limited motor.

All intentional reactivity additions, such as fuel loading, control rod loading, water flow into thermal core, etc. can be limited to less than 0.1 \$/sec.

The foregoing discussion is intended to indicate the upper limit to the rates of reactivity inputs that have even a remote probability of occurring. On this basis, consideration of several plausible sequences of events has led to the following postulated Maximum Credible Accident:

- a. By violation of procedure, the fast core has been changed unknowingly to be near critical in the shutdown condition.
- b. As water is pumped into the thermal core for a standard start up, the reactor goes critical (not possible without violation of procedures). The thermal buffer is worth about \$10 over a 20 cm region of the fast core. The water fill rate is 1.1 cm/sec. This amounts to a ramp of 0.5 \$/sec.
- c. It is assumed that period and power level scrams fail, but that the water pump continues adding water.
- d. The resulting accident is less severe than the calculated accident of \$1.0/sec ramp in the above Table. In that accident, reactor reached \$1.16 reactivity when appreciable heating began. Doppler limited the fast rise in temperature to 1200°F at 1.2 sec after reactor became critical. The U-235 fuses melted and released the fuel bundles at about 1.2 seconds. The accident is terminated by falling fuel about 0.1 seconds later, by which time the fuel temperature has risen to 1500°F.

- e. The accident size is approximately 1000 MW-sec.
- f. There is no explosion.
- g. The fuel clad does not melt (MP = 2500°F). Fission product release is limited to that of the melted fuses (amounting to less than 0.3% of the U-235 in the core). The small amount of fission product contamination is contained in cell by standard particulate filter with the exception of the inert gasses and iodine. The dispersed fission products in the cell amount to about one curie after one week.
- h. The U-235 fuse pins are at the mid-plane of the core. In this relatively fast accident, the heat generated in the pins by fission does not have time to dissipate and the temperature of the pins is related to the temperature of the oxide fuel simply by the respective enrichments and heat capacities. When the fuel reaches 500°F, the pins reach the melting temperature of uranium metal.

16.2.6 Concluding Remarks

This discussion considers only those aspects of the MSSCE hazards which may be different from hazards in the existing Critical Experiment Facility. The primary difference between the maximum credible accident for the MSSCE and the other cores is that no moderator is present to carry away heat generated by the accident. Further, there is no steady-state operating level comparable to the boiling condition in the BWR criticals. To circumvent this detail, the MSSCE is fused so that the core drops to terminate an excursion after reaching the fuse temperature. As in the BWR cores, the accident size is limited by Doppler coefficient. The analysis indicates that for the range of accidents considered, the shorter lifetime of the fast core does not contribute to the nuclear hazard. The analysis presented in this preliminary safeguards analysis indicates that Mixed Spectrum Superheater Critical Experiments can be safely carried out in APED VAL Critical Experiment Facility.

FIGURE 16.1.1 SKETCH SHOWING ELEVATION OF PROPOSED MSSR CRITICAL EXPERIMENT FACILITY.

FIGURE 16.1.2 CORE CROSS SECTION OF PROPOSED MSSR CRITICAL EXPERIMENT FACILITY

FIGURE 16.1.3 SCHEMATIC DIAGRAM OF FAST CORE FUEL ELEMENT FOR MSSR CRITICAL EXPERIMENT.

17.0 APPENDIX

I. FUEL ROD STRUCTURAL ANALYSIS

A. Design Information

1. Fabrication - The fuel rod will be fabricated utilizing the swaging process with vibratory compaction of the UO₂. It will be assumed that the swaging is accomplished cold, without an annealing process.

2. Materials Properties

a. Clad - The Hastelloy-X cladding material is assumed to have the following properties. (It is noted that the swaging process, stress corrosion and radiation damage effects on the clad material have not been factored into an evaluation of the allowable stress. When materials property data indicating the magnitude of these effects on the allowable stresses is available a more meaningful allowable stress level can be established)

	Rm. Temp.	800°F	1000°F	1200°F
Mean Coef. of Thermal Expansion ^{1*}	$\frac{\text{in}}{\text{in}^\circ\text{F}}$	7.5x10 ⁻⁶	8.2x10 ⁻⁶	8.4x10 ⁻⁶ 8.6x10 ⁻⁶
Modulus of Elasticity ¹	$\frac{\#}{\text{in.}^2}$	28.4x10 ⁶	24.4x10 ⁶	23.3x10 ⁶ 22.0x10 ⁶
0.2% Yield Strength ²	$\frac{\#}{\text{in.}^2}$		43,000	42,000 40,000
Tensile Strength ²	$\frac{\#}{\text{in.}^2}$		99,000	93,000 83,000
Density	0.28 $\frac{\#}{\text{in.}^3}$	Specific Heat 0.12 $\frac{\text{Btu}}{\#^\circ\text{F}}$, Thermal Conductivity 12 $\frac{\text{Btu}}{\text{hr ft }^\circ\text{F}}$		

* The superscripts refer to the Bibliography.

Steady State Creep relationship (an approximation derived from data³ for Hastelloy-X at 1750°F).

$$\Delta\epsilon_c = 0.855 \times 10^{-7} \sigma^{0.86} (10)^{\left[\frac{T_2}{2210} (20 + \log t) - 20 \right]}$$

where $\Delta\epsilon_c$ is the strain, T_2 temperature °R, t time hours and σ stress in psi.

- b. Nuclear Materials - The following physical characteristics were used for the nuclear materials:

376°F 1076°F 1886°F

Mean Coef. of Thermal

Expansion⁴ $\frac{\text{in}}{\text{in } ^\circ\text{F}}$ 5.0×10^{-6} 6.1×10^{-6} 7.2×10^{-6}

(values at higher temperatures were extrapolated from the above values)

Thermal Conductivity $1.3 \frac{\text{Btu}}{\text{hr ft } ^\circ\text{F}}$, Density $0.37 \frac{\text{#}}{\text{in}^3}$,

Specific Heat $0.08 \frac{\text{Btu}}{\text{# } ^\circ\text{F}}$

3. Conditions

The operating conditions (heat generation rates, flow rates and conditions, and dimensions) are those described in Section 2.5 for the operating design point.

B. Summary of Fuel Rod Structural Analysis Results

The following results when compared with the material yield values should be tempered by the uncertainties in materials allowables under actual operating conditions and the uncertainties in operating characteristics deviations. Our next phase of development for the MSS concept should produce information that will allow us to make a more accurate evaluation of the fuel rod stress picture.

Clad Stresses (psi)		σ_z	σ_r	σ_θ
1. External pressure		--	-1000	--
2. Thickness temperature gradient (+ OD, - ID)		6850	--	6850
3. Circumferential temperature gradient (+ depending on circum. location)		9600	--	--
4. Differential radial expansions		--	-1590	11,600
5. Fission gas pressure t = 0		--	--	--
(w/3" length fission gas chamber)	t = 32 Mo.	8500	--	17,000
6. Axial temperature gradient		10	--	--
7. Core radial temperature gradient (+ depending on location)		3370	--	--
8. Local spacer loading	OD	2230	--	3210
	ID	-2730	--	-4210

The equivalent (combined) stress

$$\sigma_{eq} = \frac{1}{\sqrt{2}} \left[(\sigma_r - \sigma_\theta)^2 + (\sigma_\theta - \sigma_z)^2 + (\sigma_z - \sigma_r)^2 \right]^{1/2},$$

for the combination of stresses giving the maximum, yields:

$$\sigma_{eq} = 22,200 \text{ psi} \quad @ t = 0$$

$$\sigma_{eq} = 29,500 \text{ psi} \quad @ t = 32 \text{ months}$$

An evaluation of creep under the above stress conditions over a 32 month period, using the approximate creep relationship given in section A-2-b above shows no significant creep.

C. Fuel Rod Structural Analysis

1. External Pressure - During initial operation of the fuel rod the external (differential) pressure across the clad will be approximately 1000 psi. In the case where the nuclear

material is rigidly against the clad, a radial compressive stress of 1000 psi will result in the clad. In the fission gas chamber area the clad stress (prior to fission gas buildup) will be

$$\sigma_r = \frac{PR}{t} = 6800 \text{ psi}$$

$$\sigma_z = \frac{PR}{2t} = -3200 \text{ psi}$$

The critical⁵ buckling (external) pressure for the fission gas chamber is:

$$P_{crit} = \frac{Et^3}{4R^3} = 17,500 \text{ psi} \quad (\text{assuming a metal temp. of } 1200^\circ \text{F})$$

As the fission gas pressure builds up, the fission gas chamber external pressure will be reduced.

2. Thickness Temperature Gradient - Stresses in a thin tube with a thickness temperature gradient can be approximated with the equation:⁶

$$\sigma_z = \sigma_\theta = \frac{\alpha E \Delta T}{2(1-\nu)} = 6850 \text{ psi}$$

where E = modulus of elasticity psi

α = mean coefficient of thermal expansion $\frac{\text{in.}}{\text{in.}^\circ \text{F}}$

ΔT = temperature difference between the tube ID and OD

ν = poissons ratio (0.3)

3. Circumferential Temperature Gradient - The tube circumferential temperature gradient stresses can be approximated with the equation:⁶

$$\sigma_z = E\alpha \left[T - \int_0^{2\pi} \frac{T}{\pi} d\theta \right]$$

By assuming that the circumferential clad temperature variation

is as great as the circumferential heat transfer coefficient variation³ (the actual circumferential temperature gradient should be less):

$$\left[T - \int_0^{2\pi} \frac{T d\theta}{\pi} \right] \sim 50^\circ\text{F}$$

thus $\sigma_r = \pm 9600 \text{ psi}$

4. Differential Expansion Stresses (clad to UO_2) - Based on the assumption that there will not be a clearance between the clad and the nuclear materials at room temperature, a gap or interference, as the case may be, for the design point operation will have the value:

$$\Delta R = (T_m - \frac{1}{3} \Delta T_m) \alpha_m - T_c \alpha_c$$

where T_m = max. temperature $^\circ\text{F}$

ΔT_m = meat temperature gradient $^\circ\text{F}$

T_c = clad temperature $^\circ\text{F}$

α_m, α_c = meat and clad mean coefficient of thermal expansion respectively $\frac{\text{in}}{\text{in}-^\circ\text{F}}$

ΔR Radial difference in $\frac{\text{in.}}{\text{in.}}$

Stresses in the clad corresponding to the above strain values would, assuming the solid meat does not compress, have the value:

$$\sigma = \Delta R E$$

Experience⁷ shows that in actual practice the meat density (at least as far as a unit cross-section is concerned, ignoring the fact that some material may actually move axially) is increased some 5%.

Stresses in the clad calculated without an increase in density due to compression have a maximum $\sigma_\theta = 496,000 \text{ psi}$. Assuming that the meat does compress, the maximum clad stress due to differential radial expansion is $\sigma_\theta = 11,600 \text{ psi}$.

$$\sigma_r = -1590 @ 10$$

$$\sigma_r = 0 @ 00$$

This phenomenon of meat compaction under compressive loads during operation requires further experimental evidence. A 500,000 psi stress would certainly flow, relieving the differential expansion stresses, but the higher stress levels would in all probability lead to stress corrosion problems.

5. Fission Gas Pressure - On the basis of a 100,000 MWD/Mt burnup a fission gas chamber 3" long will limit the fission gas pressure buildup⁸ at the end of 32 months to 2500 psi. This fission gas pressure buildup results in clad stresses of

$$\sigma_0 = \frac{PR}{t} = 17000 \text{ psi}$$

$$\sigma_z = \frac{PR}{t^2} = 8500 \text{ psi}$$

6. Axial Temperature Gradient - The stresses in a tube resulting from an axial temperature gradient have the form:

$$\sigma_z = \frac{E t R \alpha_c}{(1-\nu^2)} \frac{\partial^2 T}{\partial z^2}$$

Evaluating $\frac{\partial^2 T}{\partial z^2}$ based on the normal power factor curve:

$$\sigma_z < 10 \text{ psi}$$

7. Core Radial Temperature Gradient - Based on a power factor curve having a maximum radial power factor of 1.4, the maximum core radial temperature gradient across a fuel rod is $\sim 7.4^\circ\text{F}$. This gradient across the fuel rod would, if not restrained, cause a 0.013 inch deflection in a 30" long rod.

The fuel rod deflection will be restrained by the fuel rod wire wrapping. Assuming that there are 8 points of support, each support point would have a bearing stress of approximately 500 psi. Stresses in the clad resulting from the restraint of the lateral deflection are:

$$\sigma_z = \pm 3370 \text{ psi}$$

8. Local Spacer Loading - The local loadings described in 7. Core Radial Temperature Gradient above introduce stresses in the cylinder wall.

An approximate value of these local loading stresses based on curves in Ref. 10, and a local loading of 1.5 # (0.5 # calculated) yields the following stress values.

$$\begin{aligned}\sigma_z &= 2230 \text{ psi at OD} \\ \sigma_z &= -2730 \text{ psi at ID} \\ \sigma_\theta &= 3210 \text{ psi at OD} \\ \sigma_\theta &= -4210 \text{ psi at ID}\end{aligned}$$

9. Equivalent Stress - Based on an equivalent stress of the form:

$$\sigma_{eq} = \sqrt{\frac{1}{2} [(\sigma_r - \sigma_\theta)^2 + (\sigma_\theta - \sigma_z)^2 + (\sigma_z - \sigma_r)^2]}^{1/2}$$

the maximum of the possible combinations of clad stresses yields:

$$\begin{aligned}\sigma_{eq} &= 22,200 \text{ psi at } t = 0 \\ \sigma_{eq} &= 29,500 \text{ psi at } t = 32 \text{ months}\end{aligned}$$

These equivalent stress values compare with a 0.2% yield strength at 1200°F of 40,000 psi, which would, based on the above calculations, represent a margin of 1.35, or if it is assumed that 1.5 times yield can be tolerated for thermal stresses, the margin would be 2.0. At the present stage of development these margins must be tempered by the lack of information concerning the effects of stress corrosion, radiation damage and fabrication residual stresses on the allowable properties.

Further information concerning flow distribution factors, meat compaction under stress and radiation, creep of the clad under stress and radiation, and vibratory characteristics of the rods, obtained during the next phase of component developmental testing should allow us to make a more accurate estimate of the stresses in the fuel rods.

Although there is a potential for an increased fuel rod heat flux that would result in clad stresses below the clad yield strength value as we presently know it, the above listed margins should not be reduced until such a time as the uncertainties in materials allowables and system characteristics are evaluated experimentally.

D. Fuel Rod Analysis Bibliography

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II. Superheater Pressure and Support Structure Analysis

A. Design Information

1. Operating Conditions

- a. Temperature - It will be assumed that the superheater pressure structure temperatures will be equal to the saturation temperature of the boiler water (550°F). This assumption should be valid if a thermal shield is provided on the inside of the pressure wall, between the steam flow area and the pressure wall.

There will be a radiation induced heat generation rate of 650 Btu/in³-HR in the flexible side wall (adjacent to the superheater) which will, depending on the boundary conditions, change the reference metal temperature slightly.

The superheater support grid will be assumed to have a metal temperature equal to the outlet steam temperature (950°F). The support grid metal temperature will be increased slightly due to a radiation induced heat generation rate of 650 Btu/in³-HR.

- b. Pressures - The pressure drop between the boiler outlet and the superheater inlet will be less than 5 psi. The drop across the superheater during the design operation is estimated to be 65 psi. The drop across the rear (superheater outlet section) shielding is estimated to be 5 psi.

The maximum anticipated differential pressure, across the support grid and superheater outlet pressure structure walls, (resulting from a loss of coolant accident) is estimated to be 1000 psi.

In addition to pressure drop loading, the superheater support grid will be carrying a total weight load of 12,800 lbs plus its own weight (~350 lb).

2. Design Values

- a. Pressure Loading - The design pressure for normal operating conditions will be taken at 1.25 times the calculated operating pressure value.
- b. Maximum Loading - A maximum pressure short-term loading (one time operation) across the superheater support grid and the superheater outlet pressure structure will be taken as 1000 psi.

The maximum loading for the superheater inlet sections will be taken as 70 psi (the same ratio of design pressure to maximum pressure as that utilized in the superheater outlet section).

- c. Stress Allowables - The stress allowable to be used for design conditions will be 0.2% yield strength at the material temperature. In the case of thermal stresses an allowable of 1.5 x (0.2% yield strength) will be used where the combined stresses do not exceed ultimate and do not contribute to intolerable distortions. Tensile strength at the material temperature will be used for the maximum loading.

3. Materials Properties

The superheater pressure and support structure will be fabricated of Inconel-X plate. The material will be conditioned by a mill anneal at 1950°F and air-cooled, 1950°F for 2 hours and air-cooled, 1525°F for 8 hours and air-cooled, 1325°F for 16 hours and air-cooled. The materials properties for the Inconel-X plate used in the design are as follows:

	<u>Room Temp.</u>	<u>550°F</u>	<u>950°F</u>
0.2% Yield Strength ¹ psi	100,000	89,000	87,000
Tensile Strength ¹ psi	165,000	154,000	148,000
Modulus of Elasticity ¹ psi	31x10 ⁶	28.9x10 ⁶	26.9x10 ⁶
Mean Coefficient of Thermal Expansion ² in/in/°F	-	-	9.0x10 ⁻⁶
Thermal conductivity ² $\frac{\text{Btu}}{\text{ft-hr-°F}}$	7.0	7.6	12.1

4. Configuration

A cross-sectional picture of the assembled superheater pressure structure and superheater support grid is shown in figure 3.1.1.

B. Summary of Superheater Pressure and Support Structure Analysis Results

Utilization of the above Design Information (Sec. A) resulted in the following sizing of superheater pressure and support structure components:

1. Superheater Outlet Pressure Structure
 - a. 0.75 inch wall thickness, design pressure limited.
 - b. 10 inch maximum axial space between shielding blocks, maximum pressure limited.
2. Superheater Inlet Pressure Structure
 - a. 0.50 inch wall thickness, maximum pressure limited.
 - b. 0.050 inch maximum deflection under design loading.
3. Superheater pressure Structure Support Legs
 - a. 4-support legs between pressure structure and primary pressure vessel.
 - b. 36 inch length of legs. Limited on upper side (of length) by buckling under maximum pressure, on lower side by differential expansions.
 - c. 4.5" x 4.5" (outer dimensions) box structure of 0.5 inch wall thickness, fixed at the superheater pressure structure end, pinned at the primary pressure vessel end.
4. Superheater Side Flexible Wall
 - a. 0.25 inch wall thickness.
 - b. Design pressure (88 psi) limited.
 - c. 0.090 inches maximum allowable gap between superheater channels and wall.
5. Superheater Support Grid
 - a. Grid of 7 inch deep by 0.5 inch wide beams, on 4" x 4" spacing.
 - b. Maximum pressure limited (1000 psi).
 - c. 0.010 inch deflection under design pressure (88 psi).

6. Inlet Duct

- a. 0.25 inch wall thickness with a splitter in the middle of the 32 inch span.
- b. Deflection limited at design pressure.
- c. 0.025 inch deflection at design pressure (6.25 psi).

7. Outlet Transition and Pipe

- a. 0.75 inch transition wall thickness.
- b. 0.375 inch outlet pipe wall thickness.
- c. Maximum pressure (1000 psi) buckling limited.

8. Analytical and Experimental Information Required Prior to Manufacture

The above analysis results must be supplemented, prior to manufacture of the superheater pressure and support structure, with the following analytical and experimental information.

- a. Experimental verification of the superheater flexible side wall operation.
- b. Analysis of the thermal stresses in the thermal shield for the superheater outlet transition and pipe.
- c. Analysis of the transient pressures associated with a loss of coolant pipe break.
- d. Analysis of the discontinuity stresses in the superheater outlet transition between the superheater outlet pressure structure and the outlet pipe.
- e. Analysis in three dimensions, of the superheater pressure structure to evaluate the buckling characteristics of the superheater side wall corner columns and the superheater inlet pressure structure.
- f. Experimental information on the creep, corrosion and radiation characteristics of Inconel-X under the conditions to be expected in operation.
- g. Analysis of thermal stresses in the superheater support grid as a result of outlet steam temperature radial variations and heat generation.
- h. Analysis of the support discontinuities introduced by the inlet duct control rod guide tubes.

C. Structural Analysis

1. Superheater Outlet Pressure Structure

The differential pressure across the superheater outlet pressure structure will be 75 psi. The design pressure loading is 94 psi. Metal temperature for the structure will be 550°F (assuming that a thermal shield is provided in all areas exposed to the outlet steam).

The pressure structure span is 32". It is assumed that the edges are built in and that the plate length is greater than 2 times the width. The maximum stress in the plate has the value³.

$$\sigma_{\max} = \frac{-0.0829 p a^2}{h^2} \times 6$$

where a = width inches

h = plate thickness inches

p = lateral pressure psi

The minimum allowable plate thickness under design conditions is:

$$h = 0.75"$$

Deflection of the plate under design conditions is³:

$$W_{\max} = \frac{0.0277 p a^4}{E h^3}$$

where E is the modulus of elasticity.

$$W_{\max} = 0.224"$$

It is assumed that in the case of a loss of coolant accident the pressure wall structure will be supported by the superheater outlet shielding blocks. Under these conditions the maximum allowable axial span between adjacent shielding blocks will be 10 inches.

2. Superheater Inlet Pressure Structure

The differential pressure across the superheater inlet pressure structure will be 5 psi. The design pressure is

6.25 psi and the metal temperature is 550°F. The maximum pressure load is 70 psi.

Structure span is 32". Edges are built in and the length of the wall is greater than twice the width. The maximum stress has the value of 3:

$$\sigma_{max} = \frac{0.0829 p a^2 \times 6}{h^2}$$

The structure is maximum pressure limited giving a minimum allowable wall thickness of:

$$h = 0.50 \text{ inches.}$$

Utilizing the minimum allowable wall thickness and design pressure (6.25 psi) the maximum deflection in the wall is 3:

$$W_{max} = \frac{0.0277 p a^4}{E h^3}$$

$$W_{max} = 0.050 \text{ inches.}$$

3. Superheater Pressure Structure Support Legs

The superheater pressure structure will be supported by four support legs, each being built in the form (cross-section) of a square box. The superheater pressure structure end of the support legs will be rigidly attached, while the primary pressure vessel end of the legs will be pinned into position.

Under the maximum pressure loading condition (1000 psi) there will be a total load on the support legs of 1,023,000#. A maximum differential expansion between the primary pressure vessel and the superheater pressure structure (deflection of a support leg) is estimated to be 0.024".

The metal temperature of the support legs will be 550°F.

The design has 36" long legs built-up in a box structure 4.5" x 4.5" (outer dimensions) with 0.5" wall thickness.

The critical buckling compressive load on a support leg is 4:

$$p_{cr} = \frac{\pi^2 EI}{(0.7 \cdot L)^2}$$

where I is the moment of inertia and L the column length.

$$p_{cr} = 7,150,000 \text{ \#/Leg}$$

The 0.024" lateral deflection of the support leg due to differential expansion requires a lateral beam end load (from simple beam theory) of:

$$p_s = \frac{EI \cdot 0.024}{0.333 \cdot L^3}$$

$$p_s = 744\#$$

The allowable beam lateral load (from simple beam theory) has the value:

$$p_{s\ all} = \frac{I}{L^2 c} \left[\sigma_{all} - 256,000 \times 0.024 - 256,000/A \right]$$

where c is the distance from the neutral axis to the extreme fiber, and A is the cross-sectional area.

$$p_{s\ all} = 9130\#$$

Because of the uncertainty of buckling calculations, the possibility of increasing the 0.024 offset deflections as a result of fabrication and thermal distortion, and possible increased loading on a particular leg during a maximum pressure surge, it is felt that the superheater support leg sizing is not unreasonably conservative.

4. Superheater Flexible Side Wall

The calculated differential pressure across the superheater flexible side wall is 70 psi. The design pressure loading is 88 psi and the structural temperature is 550°F.

The flexible side wall has been designed to be supported by the fuel channels. Fuel channels have been designed to withstand a differential pressure of 1000 psi.

Figures A.1 and A.2 show parametric curves for the flexible plate indicating the gap limits (between the flexible side wall and the superheater fuel channels) and the distance of the first point of side wall contact (with the fuel channel) from the plate edge.

The present design utilizes a 0.25 inch thick flexible side wall plate having a maximum allowable gap of 0.090 inches and a point of contact about 7 inches from the edge of the plate.

Assuming no heat transfer across the plate surface on the superheater side of the side wall plate, a boiler side metal temperature of 550°F and a radiation induced heat generation

rate of $650 \frac{\text{Btu}}{\text{in}^3 - \text{hr}}$, the side wall plate has as linear

temperature thickness gradient of 28°F and a superimposed parabolic temperature gradient of 70°F.

The thermal stress in a clamped rectangular plate having a linear temperature thickness gradient is ⁵:

$$\sigma = \pm \frac{E\alpha\Delta T}{(1-\nu)}$$

where α is the mean coefficient of thermal expansion and ΔT the linear temperature thickness gradient.

$$\sigma = \pm 7500 \text{ psi}$$

The stress in a plate with a parabolic temperature thickness gradient is ⁵:

$$\sigma = 2/3 E\alpha\Delta T$$

$$\sigma_{\max} = 1220 \text{ psi (at faces)}$$

The total (superimposed) thermal stress in the side wall plate is:

- a. On the superheater side of the plate

$$\sigma = -8720 \text{ psi (compression)}$$

- b. On the boiler side of the plate

$$\sigma = 6280 \text{ psi (tension)}$$

The thermal stresses will reduce the bending stresses due to differential pressure in the vicinity of contact between the plate and the superheater fuel channels, but these same stresses are additive near the clamped plate edge supports.

Although they contribute to distortions, the calculated thermal stresses are well below the normally accepted value of up to 1/2 x (design allowable strength) over the design allowable strength (actually in this case the thermal stresses are only 20% of that normally allowed).

5. Superheater Support Grid

The superheater support grid has a maximum pressure loading of 1000 psi and a structural metal temperature of 950°F. The grid composed of 7" deep by 0.5" wide beams on 4" x 4" spacing (32" x 32" overall) is "simply-supported" on a ledge attached to the superheater outlet pressure structure.

A simultaneous equation solution of the support grid structure redundant moments and shears at each grid joint was carried out. (This analysis did not include the effect of differential temperature expansions.) Results (redundant moments and shears at each joint) for a 1/8th symmetric section of the grid are given in Table A.1. M_n , M_p and Q_z represent respectively the beam inplane bending moment on the beam end in inch-pounds, the beam out of plane bending moment on the beam end in inch-pounds and the vertical shear on the beam end in pounds.

The maximum bending stress occurs in member 14-15 (see Table A.1) where

$$\sigma = \frac{M c}{I}$$

$$\sigma = 76,500 \text{ psi}$$

The maximum shear stress occurs in member 5-9 (see Table A.1) where

$$\tau = \frac{Q_z}{A}$$

$$\tau = 11,831 \text{ psi}$$

The maximum lateral deflection of the support grid under the maximum pressure load is 0.098 inches. Under design loading conditions (88 psi) the superheater support grid will deflect 0.010 inches.

Both the bending stress of 76,500 psi, compared with an allowable of 148,000 psi, and the shear stress of 11,831 psi, compared with a conservative allowable shear of 44,500 psi, are conservative. During the design phase of the mixed spectrum superheater, in addition to factoring temperature variations into the structural grid analysis, the superheater support grid will be sized to approach allowable stress levels.

Buckling of the deep beam segments under the loading conditions of Table A.1 although not anticipated to be a problem, has not as yet been evaluated.

6. Inlet Duct

The inlet duct metal temperature will be 550°F. The operating pressure differential will be 5 psi and the design pressure 6.25 psi. The maximum pressure loading will be 70 psi. Width (span) of the pressure wall is 32" and the length is less than 64".

TABLE A.1

SUPERHEATER SUPPORT GRID-REDUNDANT MOMENTS AND SHEARS

Member I - J	I End			J End		
	M _n	M _p	Q _z	M _n	M _p	Q _z
1 - 2	-2651	2650	2732	-2410	-2650	5268
2 - 3	42	2087	3318	-2770	-2087	4682
3 - 4	1031	1309	3630	-2509	-1309	4370
4 - 5	1600	444	3880	-2079	-4436	4120
2 - 6	564	2378	20509	65472	-2378	-12509
3 - 7	778	1740	31927	110930	-1740	-23927
4 - 8	865	909	38867	138600	-909	-30867
5 - 9	444	0	20700	74355	0	-16700
6 - 7	-65007	1913	12509	99042	-1913	-4509
7 - 8	-98706	1203	7974	114600	-1203	26
8 - 9	-114440	40	5261	119480	-40	2739
7 - 10	-110220	1404	20461	176060	-1404	-12461
8 - 11	-137800	75	25580	224130	-75	-17580
9 - 12	-73950	0	13960	121790	0	9960
10 - 11	-175570	914	12460	209942	-914	-4461
11 - 12	-209140	292	6662	219790	-292	1338
11 - 13	-223500	467	15378	269020	-467	-7378
12 - 14	-121500	0	8623	147990	0	-4623
13 - 14	-268660	114	7377	282170	-114	623
14 - 15	-147880	0	4000	155880	0	0

Lateral Deflection at Joints

Jt. No.	Def. In.	Jt. No.	Def. In.
1	0	8	-0.036
2	0	9	-0.039
3	0	10	-0.052
4	0	11	-0.066
5	0	12	-0.071
6	-0.017	13	-0.085
7	-0.029	14	-0.092
		15	-0.098

Plate Grid and Joints

Stresses and deflections calculated using the equations³:

$$\sigma_{\max} = \frac{-0.0829 p a^2 \times 6}{h^2}$$

$$W_{\max} = \frac{0.0277 p a^4}{E h^3}$$

indicate that the inlet duct wall will be deflection limited at the design pressure and stress limited at the maximum pressure.

In order to reduce the lateral deflection a splitter was put at the center of the span (reducing the deflections by a factor of 16). A 0.25 inch wall thickness plate under design pressure results in:

$$W_{\max} = 0.025 \text{ inches}$$

$$\sigma_{\max} = 12,900 \text{ psi}$$

For the maximum pressure load (70 psi):

$$\sigma = 145,000 \text{ psi}$$

and the deflection is:

$$W = 0.280 \text{ inches}$$

Added stiffness (not taken into account) is provided in the inlet duct area by the rigid attachment of the control rod guide tubes.

7. Superheater Outlet Transition and Pipe

The superheater outlet transition and pipe will operate at a pressure differential of 75 psi with a design pressure differential of 94 psi. The maximum pressure differential will be 1000 psi. Structural metal operating temperature will be 550°F (a heat shield being provided between the structural wall and the steam).

The pipe wall thickness, determined from the critical pressure relationship⁶:

$$h = R \left[\frac{p(1-\nu^2) \times 12}{E(n^2-1)} \right]^{1/3}$$

where R is the mean radius and $n = 2$. Under maximum pressure h has the value:

$$h = 0.375 \text{ inches.}$$

Stress values for the pipe with $h = 0.375$ " are far below the allowable values.

The transition between the superheater outlet pressure structure and the outlet pipe will be made with the same wall thickness as the superheater outlet pressure structure wall (0.75 inches). Discontinuity stresses at the superheater outlet pressure structure--outlet transition intersection, and the outlet transition--outlet pipe intersection, have as yet to be calculated.

There will, providing the transition heat shield is attached rigidly to the outlet pressure structure, be a thermal stress induced at the point of attachment of the heat shield. This thermal stress has not as yet been calculated.

D. Superheat Structure Bibliography

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III. SUPERHEATER EXIT VENTURI ANALYSIS

A flow-restricting venturi at the steam outlet was incorporated into the design of the 300 MWe MSSR reference design. The purpose of the venturi was to limit the mass flow rate, hence pressure drop, across the fast core in the event that the main steam line should rupture. This was done by sizing the venturi such that the flow would become choked when the mass flow rate of steam was increased by 50 per cent over normal steady-state flow.

After considerable study and weighing of advantages and disadvantages, it was decided to omit the venturi from the design of the 75 MWe prototype plant. The reasons for this decision are discussed below:

1. The venturi has just one purpose, i.e., to prevent collapsing of the fast core by limiting the rate at which the reactor would blow down if the main steam line should undergo a sudden and complete rupture. The probability of this occurrence is very remote.
2. Pressure recovery is good, but not perfect, so that the venturi adds about 15 psi of pressure drop.
3. The steam velocity in the throat under normal full-power operation is of the order of 900 ft/sec. Considerable erosion can be expected, which would necessitate periodic changes and maintenance of the venturi liner.
4. Design of the exit line to permit access to the venturi proved to be less desirable than designing the fast core and its supports to withstand the full reactor pressure.

The determining factor in deciding to eliminate the venturi was the final item listed above. An analysis of the fast core, grid, and support structure showed that it was feasible to design these components to withstand the full reactor pressure of 1000 psi. This appeared to be the more desirable approach, since the venturi would penalize reactor performance somewhat. It should be noted that the safety of the system is not compromised because reasonable assurance can be provided that the fast core would not collapse if the main steam line should fail. The analysis of the fast core support structure is given in Section II of this Appendix.

As an indication of the effectiveness of the venturi in limiting the blow-down rate, the pressure decay in the reactor vessel was calculated for two cases. The first case had a venturi (5.37 inch diameter throat) in the steam exit line. The second case was simply a nozzle leading to an 18 inch, schedule 120 pipe. In the calculations, it was assumed that the pipe break occurred at normal full power operating conditions. Only steam was considered to flow through the nozzle; two-phase flow was not considered. The results are therefore valid only for comparison between the two cases, since the absolute magnitudes may be seriously in error.

The results of the calculations are shown in Figure A.3. It can be seen that with the venturi the pressure would decay to about 400 psi in 30 seconds. Without the venturi, the pressure would decay at about 75 psi in the same time interval. In the latter case, a higher percentage of the total pressure drop is taken by the fast core. On the basis of the assumptions used here, the maximum pressure is estimated to be 750 psi. In a simple pipe rupture, unaccompanied by a simultaneous nuclear excursion, this pressure should not exceed the 1000 psi for which the core structures are designed.

IV. FAST CORE TRANSIENT TEMPERATURE CALCULATIONS

A. Introduction

For the Mixed Spectrum Superheating Reactor, it is imperative that adequate cooling for the fast core be available at all times. Under normal conditions, cooling is provided by steam flow, which can be maintained by flowing either through the turbine or through a full-flow by-pass line to the condenser. For emergency conditions, however, in which containment isolation is required, the condenser heat sink is not available. Other means must be provided to cool the reactor under these conditions.

The alternate means of cooling, to be effective, must be put into operation before any part of the fast fuel cladding can reach its melting point. To provide a basis for estimating the times available for putting emergency cooling into operation, a computer program was carried out to calculate the temperature history of the cladding for the three conditions described below.

B. Cases Considered

Three cases are considered in the calculations. These are designated as Cases A, B, and C respectively. In each case, the initial conditions are those corresponding to normal steady-state, full-power operation. Time zero corresponds to the time at which the coolant conditions begin to change. The three cases are as follows:

Case A: Coolant is lost, and reactor fails to scram. Power is maintained at 100 per cent of full power. No cooling is assumed after time zero.

Case B: Coolant flow goes from 100 per cent at time zero to 10 per cent in 0.75 seconds. Reactor scrams at 0.500 seconds, and power decays as follows:

<u>Time (sec)</u>	<u>Power (% of full power)</u>
0.0	100
0.5	100
0.75	79.5
1.5	18.0
3.0	16.0
6.5	6.0

Linear interpolation is used for the intervals between the points listed above. The heat-transfer coefficient

is assumed to be given by

$$h = 1280 \left(\frac{\% \text{ flow}}{100} \right)^{0.8} \frac{\text{Btu}}{\text{hr ft}^2 \text{ } ^\circ\text{F}}$$

Case C: At time zero, reactor power drops to 6 per cent of full power, and flow instantaneously reduces to 50 per cent of full flow.

C. Computing Constants

1. Fuel parameters

	<u>Meat</u>	<u>Clad</u>
Cp, Btu/lb $^\circ\text{F}$	0.08	0.12
k, Btu/hr ft $^\circ\text{F}$	1.3	12.58
ρ , lb/in ³	0.37	0.28

2. Steady-state full flow, 232 lb/hr per rod

3. Fuel-clad gap coefficient, 500 Btu/hr ft² $^\circ\text{F}$

4. Steam coolant properties library (use linear interpolation)

<u>Temperature, $^\circ\text{F}$</u>	<u>Cp, Btu/lb $^\circ\text{F}$</u>	<u>ρ, lbs/ft³</u>
540	1.34	2.286
550	1.20	2.206
560	1.09	2.142
580	0.98	2.035
600	0.89	1.946
640	0.78	1.803
680	0.70	1.691
700	0.68	1.643
760	0.63	1.522
800	0.61	1.454
900	0.58	1.315
1000	0.57	1.206
1200	0.56	1.040
1600	0.58	0.823
2000	0.59	0.714

D. Discussion of Results

Calculations describing the temperature history of the hottest cladding point in the fast core were made for the three cases listed above. The results are shown graphically in Figure A.4. It can be seen from the figure (Curve A) that even for the most extreme case, where full power is maintained and no coolant is available, that three seconds must elapse before the clad will begin to melt. This appears to be adequate for putting emergency cooling into operation.

In Case B, where the flow decays to 10 per cent of full flow in 0.75 seconds, and the power varies according to the fission decay heat rate, the clad will begin to melt in about 5 seconds. In making the calculation, it became evident that if the flow decays at a significantly faster rate than the heat generation rate, as it does for this case, the initial portion of the temperature history will be relatively independent of the flow. This is explained by the fact that if the flow decays faster than the heat-generation rate, very little heat is removed by the coolant, and almost all of it goes toward heating the fuel and cladding. The maximum temperature attained and the equilibrium temperature are, of course, directly influenced by the coolant flow rate.

Case C corresponds to a sudden closure of the turbine admission valves accompanied by a simultaneous failure of the by-pass valves. Steam then flows into the emergency condensing tank for approximately 20 seconds before the emergency condenser goes into operation. The condensing tank is sized to take 50 per cent of full flow for 20 seconds.

Curve C shows that the maximum clad temperature will rise about 170°F above its steady-state operating temperature during the transient. It should be noted that the assumption that both coolant and heat generation undergo a step change at time zero represents an approximation to the actual case. The two assumptions are compensating, however, so that the maximum temperature attained should not be greatly different from that indicated by Curve C. If a time delay in scrambling is included, the initial slope of Curve C would be the same as for Curves A and B. Since the flow levels off at 50 per cent, while the power decays to 18 per cent in one second after scram, the temperature peak should occur in the same order of time as that indicated by Curve C. A more exact calculation incorporating both flow and power decay rates will be carried out to check this point when the equipment is to be designed.

Figure A.5 shows the temperature variation of the fuel meat for the same three conditions. In Case A, with no cooling, the UO_2 will reach its melting temperature (4800°F) in about 5.0 seconds. In the other two cases, where the reactor scrams and partial cooling is maintained, the meat temperature does not rise above its steady-state value. Thus, it can be concluded that capability for preventing the fuel from melting, which is important for a fast reactor, can be designed into the plant. The adequacy of specific items provided for this purpose must be evaluated as the design is carried out.

FIGURE A.1 SUPERHEAT CORE SHROUD PARAMETER CURVES SHEET 1

FIGURE A.2 SUPERHEAT CORE SHROUD PARAMETER CURVES SHEET 2

FIGURE A.3 PRESSURE DECAY IN STEAM LINE FOLLOWING STEAM LINE RUPTURE

FIGURE A.4. MAXIMUM CLADDING TEMPERATURE AS A FUNCTION OF TIME FOR THREE TRANSIENT CONDITIONS.

FIGURE A.5. MAXIMUM FUEL TEMPERATURE AS A FUNCTION OF TIME FOR THREE TRANSIENT CONDITIONS.