

GEAP-3589(Rev. 1)

STEAM-COOLED POWER REACTOR EVALUATION FOR 300 MW(e) SEPARATE SUPERHEATER REACTOR

November 1960

Atomic Power Equipment Department
General Electric Company
San Jose, California

metadc101056

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STEAM-COOLED POWER
REACTOR EVALUATION
FOR 300 MW(e) SEPARATE SUPERHEATER REACTOR

November, 1960

Prepared For
THE U. S. ATOMIC ENERGY COMMISSION

Under

CONTRACT NO. AT(04-3)-189, PROJECT AGREEMENT 13

Atomic Power Equipment Department
GENERAL ELECTRIC COMPANY
San Jose, California

May 11, 1961

ADDENDUM

The superheater described in this report utilizes fuel with zirconium clad in the boiling part of the plant. In practice the choice of cladding will, of course, be made on the basis of reliability and economic conditions at the time of fuel fabrication. An analysis of the power costs based on the economic ground rules provided by the AEC for this study indicated that the use of thin clad (11 mil thickness) stainless steel would result in improved economy.

As compared to zirconium, thin clad stainless steel requires increased enrichment which results in higher depletion and inventory charges. However, this is offset by a decrease in fabrication and first core capitalization costs. The net effect of the change from zirconium to thin stainless steel cladding is a savings of slightly more than .1 mill/kw hr in total power costs. The economic analysis performed by the AEC is based on the use of thin stainless steel cladding in the boiling core.

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0.0 Introduction

0.1 Scope

This report provides the preliminary design for the nuclear portion of a 300 MWe Boiling Water Separate Superheater Plant to be in operation by 1967. The research and development programs required to insure plant feasibility and the construction schedule also are presented. In addition, the potential for the Boiling Water Separate Superheater concept which may be realized in a plant to be in operation in 1975 is assessed in terms of projected plant capacity, technical improvements and potential for cost reduction.

Economic considerations have established that no incentive exists for a Boiling Water Separate Superheater in the 40 MWe size; therefore, it has been excluded from this study.

The scope of work has been limited to the accumulation of design, performance, and cost information on the nuclear portion of the plant in accordance with AEC directive. The technical basis for the superheat reactor design and performance is derived from work performed under Task A of the Nuclear Superheat Project under Contract AT(04-3)-189, P.A. #13.

0.2 Ground Rules for Evaluating Steam Cooled Reactor Plants

The AEC has established certain ground rules to be followed by the individual contractors in establishing reactor design criterion and for the purpose of estimating costs. The following summarizes those portions of the ground rules which are of particular importance in establishing the design basis.

- (1) Present energy costs are based on those attainable from a reactor plant which could be in operation by June, 1967.
- (2) The 1967 plant net rating is 300,000 KW.
- (3) Fuel costs for the 1967 plant are based on an equilibrium core; the equilibrium core must be attained within five years of plant start-up.
- (4) Potential energy costs are based on those attainable from a reactor plant which could be in operation by June, 1975 assuming all research and development planned under research and development programs will be successful.
- (5) Maximum superheater fuel cladding surface temperatures were to be:

<u>Cladding</u>	<u>Maximum Surface Temperature °F</u>	
SS	1250 F	maximum hot spot temp. during steady state operation
Zr	700 F	

- (6) Plant availability was to be 90% with a plant factor of 80%.
- (7) . Maximum UO_2 fuel centerline temperature using an average thermal conductivity of 1 Btu/hr-ft- $^{\circ}\text{F}$ was to be 4500°F .
- (8) Primary system piping is stainless steel.
- (9) Maximum permissible integrated Nvt at the pressure vessel, considering neutrons 1 Mev and above, was to be no greater than 10^{19} with a thirty year life at 80% plant factor.
- (10) For the potential reactor, no restriction was placed on cladding material but for the near term plant design cladding was restricted to stainless steel, inconel, or zircaloy.
- (11) The pressure vessel inside diameter was to be no greater than 13 ft.
- (12) Steam velocity was to be optional.
- (13) In considering the construction schedule for the 300 MW plant, the designer was to assume a 36 month construction period preceded by 9 months of design. For the 40 MW plant, he was to assume a 24 month construction period preceded by 9 months of design.
- (14) The turbine generator could be located either inside or outside the containment shell, depending on the designer's judgment. In the design of the 300 MW plant, multiple reactor units could be used providing total net power generation on range of 300 MWe. For the current technology plant, designers were to assume that if one reactor has maximum credible accident other reactors would be equally affected. For the potential plant, designers were to assume that if one of the multiple reactors should develop a maximum credible accident, it would have no effect on the other reactors.
- (15) No limit was set on steam pressure and temperature.
- (16) Cladding thickness was made optional.
- (17) For concepts utilizing feedwater return to the reactor, a full flow demineralizer will be used.
- (18) No limit will be set on flooding or unflooding coefficient of reactivity.
- (19) Radioactive waste disposal systems should be designed such that a plant can operate with pin hole leaks in 0.3% of the fuel elements at rated operating conditions.
- (20) Non-freestanding superheater fuel cladding may be used.
- (21) For the purposes of estimating the 1967 plant fuel cycle costs, the maximum exposure of stainless steel clad, UO_2 fuel elements will be 35,000 MWD/MTU based on the highest exposure of 1% of the fuel volume when burnup is achieved. Maximum fuel exposure is defined as the

product of the integrated maximum power/average power ratio and the average fuel exposure. The maximum power/average power ratio should be based on the equilibrium fuel cycle.

- (22) Thorium is excluded from consideration as a fuel
- (23) Either pressure containment or vapor suppression containment may be used.

0.3 Basis of Ground Rules for the Separate Superheater Reactor

The ground rules presented above consist of a summary of the more important and salient points condensed from the more general formulation of ground rules generated at Nuclear Superheat Study Meetings held on August 16, October 11-12, and November 4, 1960. All the subsequent amendments were acknowledged and incorporated along with the original list of rules in the compilation of the above list of abbreviated ground rules.

1.0 Plant General Description

In the proposed nuclear power plant, a forced circulation single cycle boiling water reactor and a separate superheating reactor combine to deliver 2,600,000 pounds of steam at 900°F and 965 psia to a tandem compound double flow turbine-generator set. The gross electrical output of this plant is 316,000 kw, the net electrical output is approximately 307,000 kw.

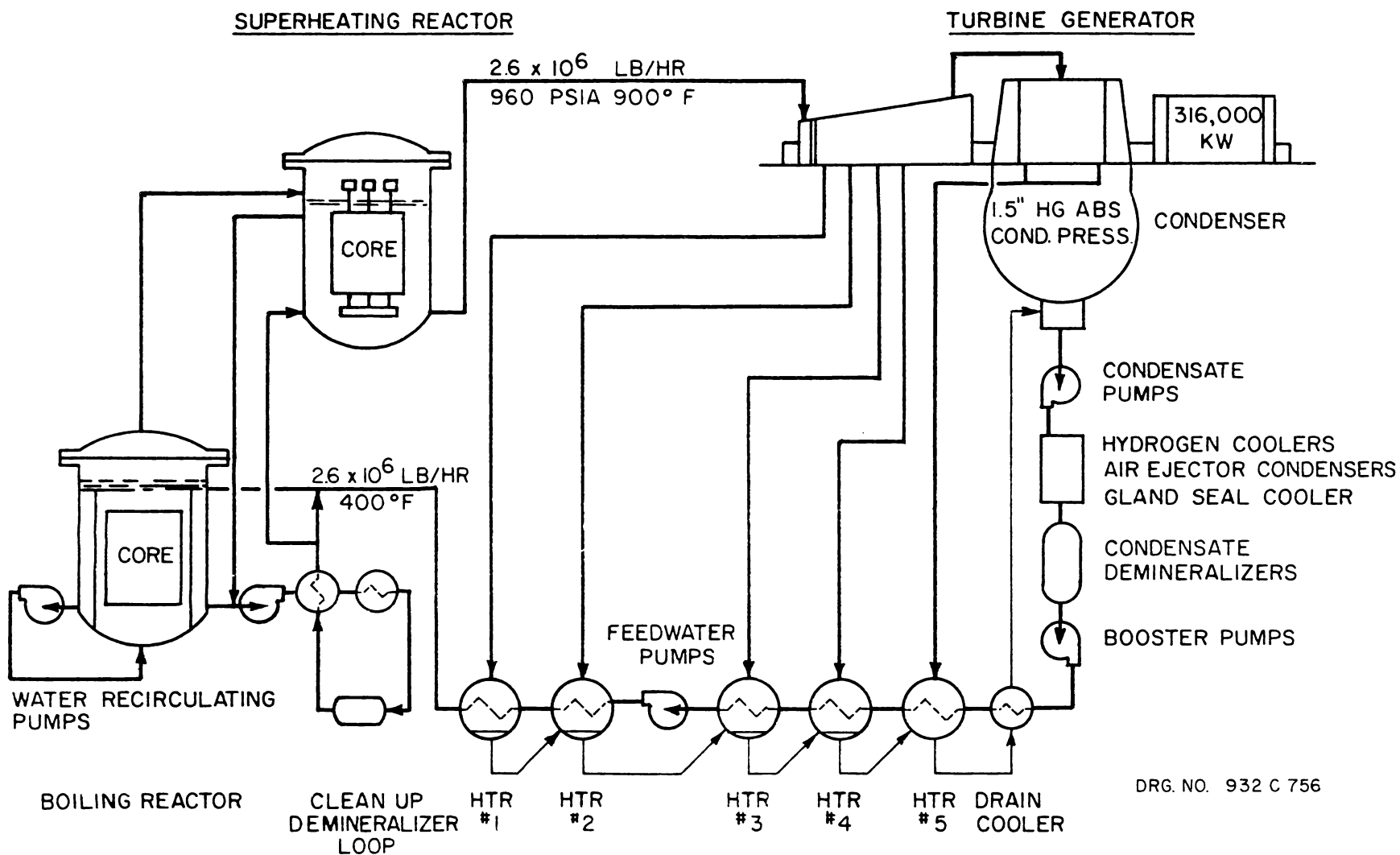
The required steam output is generated at saturated conditions in a single cycle forced circulation boiling water reactor and the steam is superheated in a superheat reactor in a separate pressure vessel. This physical separation of two different reactors permits the design of each reactor for its own conditions and limitations. The boiling water reactor is not hindered by superheat reactor limitations and the superheat reactor is not hindered by boiling water reactor limitations. The net result should be that these two reactors have the lowest fuel cost of any light-water moderated thermal reactors.

The steam flows from the superheater outlet directly to the turbine inlet. The turbine is assumed to be a tandem compound double flow unit. The low pressure turbine exhausts to a surface condenser equipped with deaerating hotwell. From the condenser, the condensate is pumped through full flow condensate demineralizers, the steam jet air ejector condenser and gland seal condenser to the feedwater heaters. From the heaters, the feedwater is pumped back to the reactors. The gross heat rate for this plant is 8997 Btu/Kw-Hr.

Design of the containment for the nuclear system is based on the pressure suppression concept. The reactors are contained in individual drywells with provision for venting of those drywells to a water pool in the remote instance of a nuclear accident resulting in a system rupture. Except for the drywell structure, the reactor enclosure is not required to see any appreciable internal pressures and is generally of conventional construction. Balance of plant structures are also conventional except for shielding requirements in certain areas.

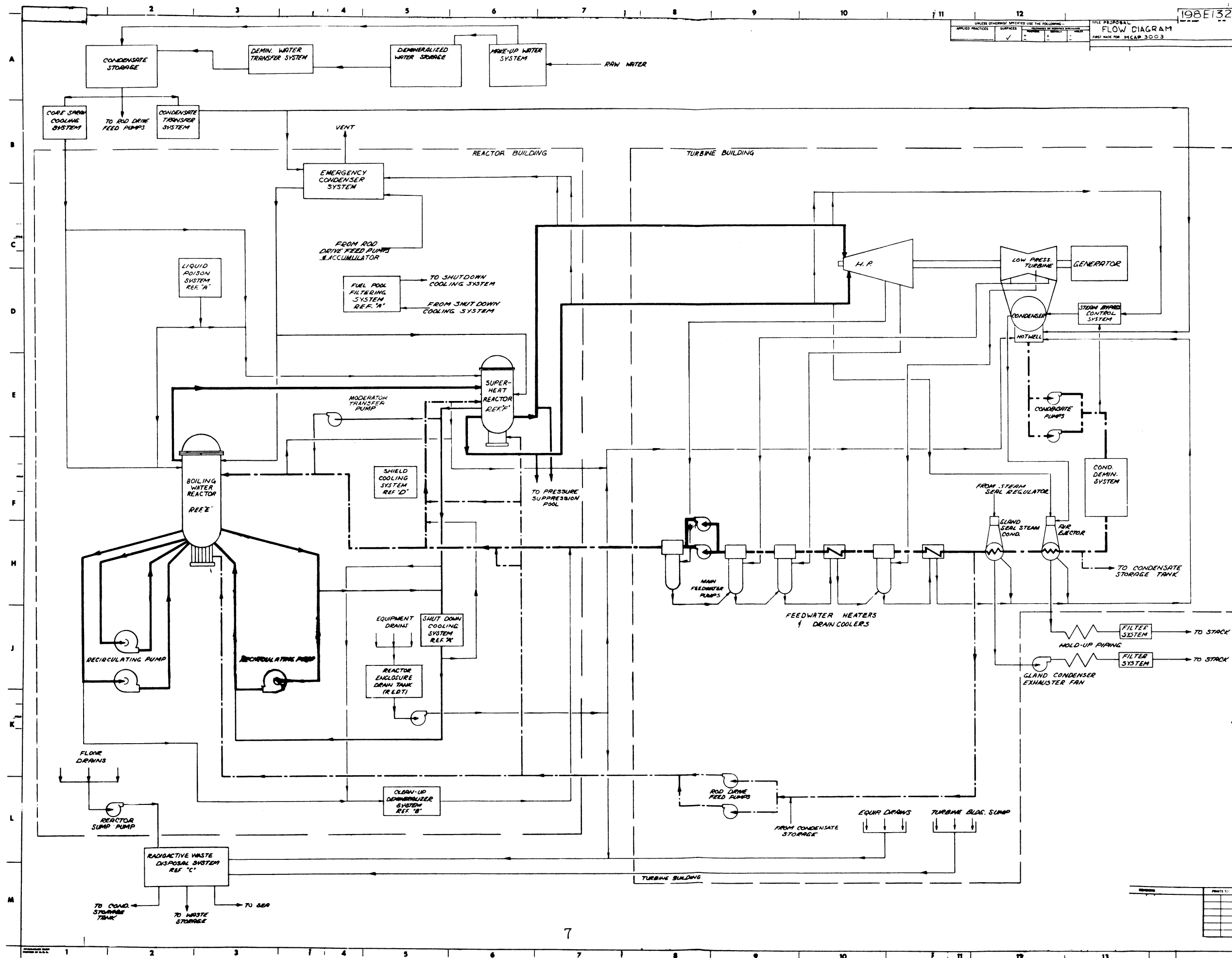
1.1 SUMMARY DATA FOR POWER PLANT

A. Gross Electrical Power	316,000 KW
B. Gross Plant Heat Rate	8997 Btu/KWH
C. Turbine	
1. Turbine Throttle Flow	2,600,000 lbs/hr
2. Turbine Throttle Pressure	965 psia
3. Turbine Throttle Temperature	900°F
4. Type	TCDF
5. Speed	1,800 rpm
6. Last Stage Bucket Size	43 in.
D. Condenser and Feedwater System	
1. Condenser Pressure	1-1/2" Hg
2. Condenser Steam Flow	1,893,000 lbs/hr
3. Number of Feedwater Heaters	5
4. Feedwater Temperature	400°F
E. Boiling Water Reactor	
1. Thermal Rating	619.2 MWT
2. Fuel	UO ₂
3. Cladding	Zircaloy
4. Moderator	Light Water
5. Core Power Density	1385 KWT/ft ³
6. Fuel Power Density	20 KW/Kg-U
Complete Boiling Water Reactor Data Sheet	Section 2.1.9
F. Superheat Reactor	
1. Thermal Rating	213.8 MWT
2. Fuel	UO ₂
3. Cladding	Stainless Steel
4. Moderator	Light Water
5. Core Power Density	1070 KW/Ft ³
6. Fuel Power Density	20.0 KW/Kg-U
Complete Superheat Reactor Data Sheet	Section 2.2.6
G. Containment	
1. Design Criteria	Rupture of 22 inch steam or water pipes
2. Type	Pressure Suppression
3. Primary Loop Coolant Inventory	200,000 lbs
4. Geometry	Drawing 198E124
5. Dimensions	Drawing 198E124
6. Design Pressure	100 psia
7. Material	ASTM-300 A201-Grade B



DRG. NO. 932 C 756

**300 MW(e) SEPARATE SUPERHEATER
REACTOR PLANT
FLOW DIAGRAM**



2.0 Nuclear Steam Supply

2.1 Boiling Water Reactor

2.1.1 General

Saturated steam required for this plant is produced in a single cycle, forced circulation boiling water reactor. This reactor incorporates significant advancement over previous designs and is based on operating experience obtained at the Dresden Nuclear Power Station and on the results of research and development activities conducted by the General Electric Company at its San Jose facilities.

The boiling water reactor design conditions are as follows:

Total thermal power	619.2 MW(t)
Steam Pressure	1075 psia
Fuel-cladding	UO ₂ - Zirc.
Fuel Specific Power	20 KW/KgU
Core Power Density	48.6 KW/liter

A complete boiling water reactor data sheet is included in Sec. 2.1.9.

2.1.2 Vessel

The reactor vessel supports and contains the nuclear reactor core, and supplies the necessary flow paths for fluids entering the core and steam and water leaving the core.

The reactor vessel is designed, fabricated, tested and stamped in accordance with the ASME Boiler and Pressure Vessel Code, Section I, and the latest applicable addenda. The latest case interpretations concerning nuclear reactor vessels are considered in the design. The corrosion-resistant stainless steel cladding inside the reactor vessel is in accordance with Section VIII of the ASME Boiler and Pressure Vessel Code.

Where applicable codes do not cover the design, detailed structural analyses are prepared to assure that the design is adequate and complies with the intent of the code.

All welded joints, including those for nozzle attachment, are designed and radiographed in accord with ASME Code standards. Joints are designed to minimize stress concentrations at changes in sections.

A stainless steel thermal shield and a water annulus between the reactor core and vessel are provided to minimize radiation damage to the vessel. The reflector and thermal shield were chosen to limit the integrated nvt at the vessel wall to less than 10^{19} nvt, based on neutrons above 1 Mev, 30 year life, and 80% load factor. The reflector thickness is 13 in.

The head closure is designed for easy removal and reassembly. A double seal is provided so that the area between the seals can be monitored for leakage. Seal welding of the top head is not required.

Removable bushings are furnished in the body flange to facilitate repair of damaged threads.

Penetrations in the lower head and flanged thimbles are provided for mounting the control rod drives. The drives themselves are bolted to, and are contained completely within, the thimbles.

The vessel is supported on pads attached to the bottom head. Lateral stability for the top of the vessel is provided by matched sets of guides or stabilizers attached to the dry well. These guides allow axial and radial vessel expansion.

Steam outlet lines are permanently attached to nozzles in the vessel body. This avoids breaking flanged joints in steam lines for head removal, making refueling easier.

Nozzles are provided on the vessel head for mounting of safety valves.

2.1.3 Internal Structure

Within the reactor vessel, the main structures are supported by the core support cylinder. This cylinder is bolted to lugs welded to the vessel near the bottom head.

Attached to the bottom of the core support cylinder is a flow baffle which serves to channel the recirculation flow to the bottom of the vessel.

Mounted on top of the core support cylinder is the core support grid. This grid is a fabricated stainless steel structure which supports and aligns the fuel channels. Also mounted on the core support cylinder is a cylindrical shroud assembly which serves to isolate the core from the downcomer annulus, and to provide upper support for the fuel channels.

Fuel channels, semi-permanently attached to the grid structure by spring clips, provide support for the fuel bundles, and assure proper distribution of coolant through the core. In addition, the fuel channels serve to guide the control rods in the core. Initial fuel channels are of stainless steel to provide reactivity control in the clean-cold condition. After initial operation, these units will be replaced with zircalloy channels.

Cruciform-shaped control rods, coupled to bottom-mounted drive mechanisms, pass through the core support grid and between the fuel channels. Control rods are fabricated of stainless steel enriched with natural boron. Vertical alignment is provided by a system of rollers in the rods and fuel channels which prevent binding during insertion and withdrawal.

Above the core, bolted to the core shroud, is a double tube sheet assembly which supports the steam separators, and also serves as a core spray and liquid poison sparger. The spray and liquid poison are discussed in later sections of this study.

The steam dryer and dry box assembly, located near the top head, complete the core assembly. This is an integral unit bolted to the vessel wall which provides final separation of steam and water and channels the dry steam to the outlet nozzles.

Feedwater return is through a sparger ring bolted to the vessel in the area below the dry box and in the downcomer annulus.

All components mounted above the core are designed for easy removal for refueling.

2.1.4 Steam Separators and Dryers

2.1.4.1 Primary Separation

The primary steam water separators for this plant are mounted on the plenum top plate above the core inside the reactor pressure vessel. The entire core flow passes through the primary units to provide a leaving steam quality of 6 percent or less water by weight and water to the downcomer at about 0.1 percent or less steam carryunder by weight. The steam separators are radial flow separators developed under the Nuclear Superheat Project. Performance data for these separators are reported in Progress Reports for that project.

2.1.4.2 Steam Drying

Steam dryers are mounted in the reactor pressure vessel head. These dryers are of the impingement type and take the steam of up to 6 percent water by weight discharging from the primary separators and dry it to 0.1 percent or less water by weight. Drying devices to accomplish this have been developed and are presently available.

2.1.5 Recirculating Pumps

The recirculation system consists of three loops, each of which contains a high capacity recirculating pump. These units take suction from the bottom of the downcomer annulus, and return to the inlet side of the core. Appropriate baffling in the reactor vessel serves to prevent mixing of inlet and outlet flows.

To protect the reactor in the event of an electrical power failure, the pumps are provided with large flywheels which insure that adequate circulation can be maintained until either power is restored or the emergency condenser circuit has been placed in operation.

The pumps proposed are vertical centrifugal units with mechanical, controlled-leakage, shaft seals and are sized for full flow start-up on cold reactor water.

2.1.6 Control Rod Drives

Each reactor control rod is individually driven by a locking piston drive mounted on the bottom head of the reactor vessel. The locking piston drive is a hydraulically operated unit initially developed for the Dresden Station. Based on testing and operations at Dresden, the current models have incorporated significant design improvements for enhanced reliability and ease of maintenance.

The drive is basically a piston operating in a hydraulic cylinder. Feedwater pressure applied to the piston drives the control rod into or out of the core. Speed of insertion or withdrawal is controlled by a flow control device in the hydraulic circuit.

The piston is contained in an annular space between concentric cylinders. The inner cylinder, fixed to the drive mechanism flange, contains magnetic position-indicating switches. The ratchet-locking device is located in the upper end of the outer-fixed cylinder.

The drive mechanisms are mounted to thimbles in the bottom head of the reactor vessel. The lower end of each thimble is flanged for mounting the drive mechanism and attaching the hydraulic pipes. The drive mechanism extends through the thimble into the reactor vessel. No rotary or linear shaft seals are required at the vessel penetrations as the drive housing is at reactor pressure. The drive housing seal to the vessel is made by flanged joints using metal "o" rings.

A coupling attached to the bottom of the control rod locks with the spud at the top of the piston tube. The coupling is devised to allow control rod replacement from above the reactor during refueling. It is also possible to uncouple the rod from below the reactor for drive removal. In this case, the control rod is back-seated against an internal extension of the vessel thimble. This seals off the thimble, and prevents draining the water from the reactor. The combination of backseating and possible uncoupling from below the reactor vessel allows the removal of a control rod drive without removing the reactor vessel head.

2.1.7 Core Design

2.1.7.1 Core Design Criteria

The boiling water core design was based on the following criteria:

- (1) The maximum fuel temperature under transient over-power conditions is to be less than 4500°F.
- (2) The minimum burnout ratio shall be 1.5 (the minimum burnout factor is defined as the minimum ratio of heat flux to cause burnout divided by the actual heat flux in a hot channel) at 125% power.

- (3) The water-to-fuel ratio is to be low enough so that the void coefficient is always negative.
- (4) The core shall possess both nuclear and hydrodynamic stability.
- (5) Within the limitations of criteria (1) through (4), the core should be designed for minimum power cost.

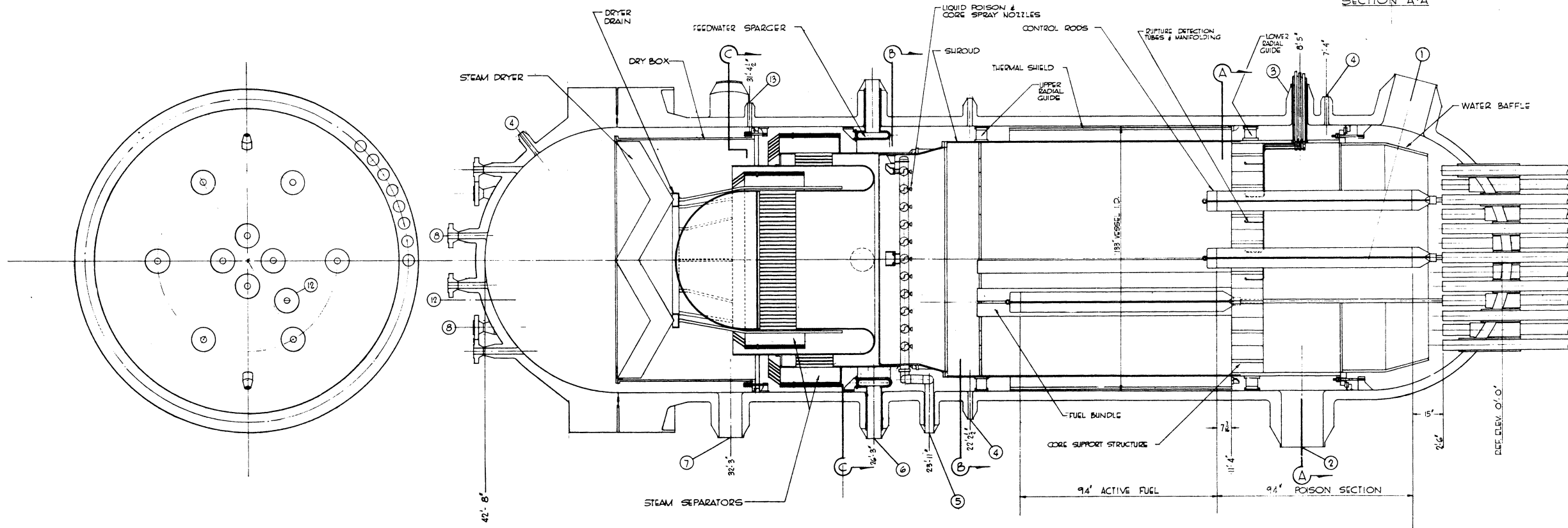
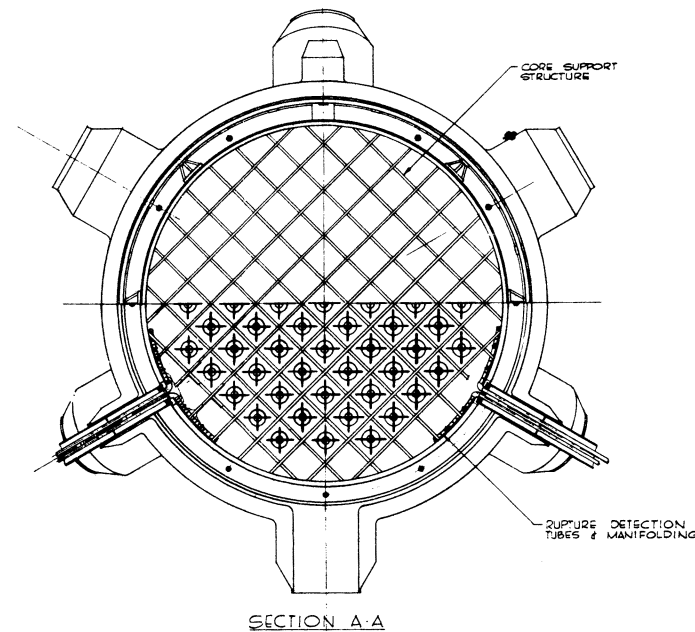
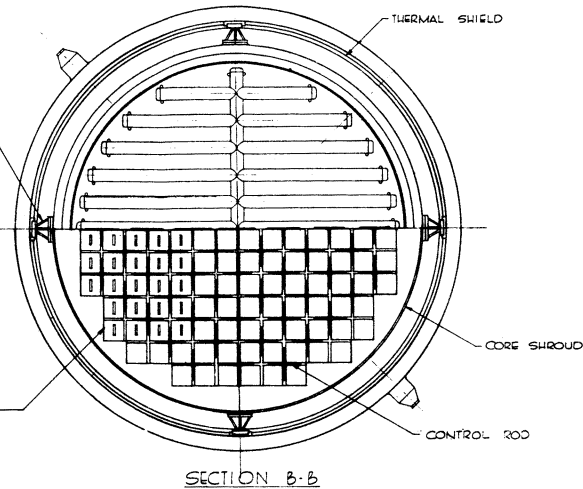
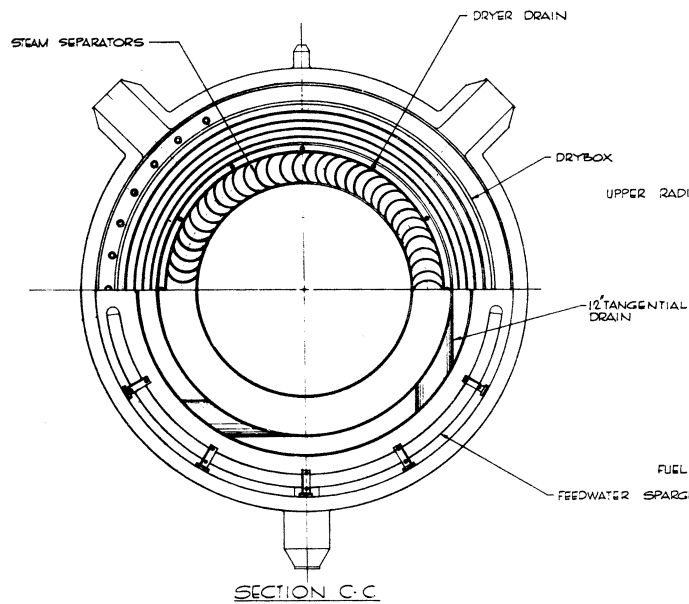
2.1.8 Fuel Assembly

The boiling water fuel assembly is shown on Drawing 198E510. The fuel assembly is a 10 by 10 array of Zircaloy clad, UO_2 cylindrical rods. Twenty corner rods have reduced enrichments and smaller fuel diameters in order to reduce water peaking effects.

A. Total Thermal Power	619.2 MW(t)	
B. Steam Conditions		
1) Steam Temperature	553.4°F	
2) Steam Pressure	1075 psia	
3) Steam Flow	2,564,000 #/hr	
C. Reactor Description		
1) Reactor Vessel		
a) Inside Diameter	11.08 ft.	
b) Inside Height	42.00 ft.	
c) Wall thickness - base metal	5.0 in.	
wall cladding	0.24 in.	
total wall	5.25 in.	
d) Material - base metal	ASTM-A-302 Grade B	
wall clad	ASTM-A-264 Grade B	
	SS type 304-modified	
e) Design Pressure	1310 psig	
f) Design Temperature	650°F	
2) Reactor Core		
a) Equivalent Diameter	8.50 ft.	
b) Active Height	7.83 ft.	
c) Active Core Volume	445 ft. ³	
d) Total Uranium Loading	30,800 Kg	
Uranium Loading-Standard Rods	26,900 Kg	
Uranium Loading-Corner Rods	3,900 Kg	
e) Average Uranium Loading	1.39% a%	
Average Uranium Loading-Standard Rods	1.45% a%	
Average Uranium Loading-Corner Rods	0.96% a%	
Uranium Loading Change with Life	Equilibrium core	
	Standard Rods	Corner Rod
Initial enrichment	2.30 a%	1.80 a%
Middle of Life	1.45 a%	.96 a%
Final Enrichment	0.89 a%	0.52 a%
Plutonium at end of life	0.75 a%	0.80 a%
f) Structural Material		Zircaloy
g) Neutron Moderator		Light Water
h) Moderator to Fuel Ratio		2.2
3) Reflector		
a) Material		Light Water
b) Axial Thickness, ft.		8 ft.
c) Radial Thickness based on equivalent diam.		13 in.
4) Fuel Elements		
a) Fuel Material		UO ₂
b) Fuel Element Geometry		Solid Rod
c) Clad Material		Zircaloy
d) Fuel "Meat" Diameter		
Standard Rods		0.352 in.
Corner Rods		0.332 in.

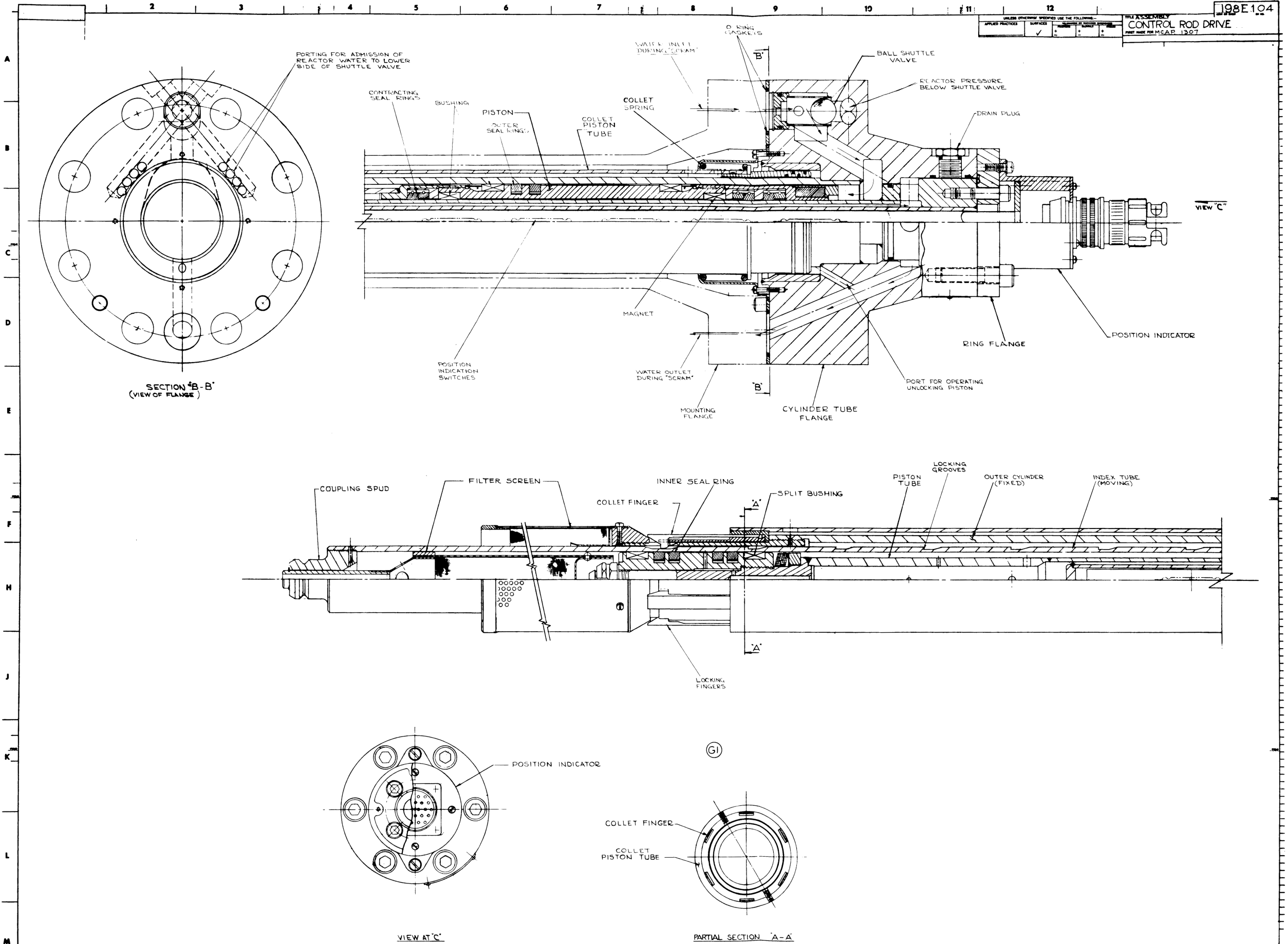
e) Clad Thickness	.025 in.
f) Fuel Clad Gap (Cold)	None
g) Gap Material	He
5) Fuel Assemblies	
a) Total Number	164
b) Number of Rod Elements per Assembly	143
Standard Rods	123
Corner Rods	20
c) Cross Sectional Dimension	6.387
d) Lattice Spacing	7.07
e) End Fitting Materials	Zircaloy
6) Reactor Control	
a) Method of Control	Control Rod Movement
b) Absorber Material	Boron Steel
c) Number of Control Elements	69
d) Cross Sectional Dimension	11 in. x 11 in.
	Cruciform
e) Effective Length	7.83 ft.
f) Type of Drive	Hydraulic
	Locking Piston
D. Performance Data	
1) Reactor Coolant	Light Water
2) Reactor Coolant Outlet Temp.	553.4°F
3) Reactor Coolant Inlet Temp.	534.8°F
4) Primary System Operating Pressure	1060 psig
5) Primary Coolant flow	25,600,000 #/hr
Average Exit Quality	10%
6) Average Core Coolant Velocity	
Entrance	4.5 ft/sec
Exit	10.8 ft/sec
7) Max. Fuel Temperature	4500°F
8) Max. Cladding Temperature	580°F
9) Burnout Margin	1.5
10) Maximum Core Heat Flux	
125% Power, 96% of heat in fuel	392,000
11) Average Core Heat Flux	107,000
12) Average Core Power Density	1385 KW/ft ³
13) Peak-to-Average Power Ratio	
Axial	1.60
Radial	1.25
Inter-Control Rod	1.20
Local	1.22
Overpower	1.25
Total	3.66
14) Average Specific Power	20.0 KW/Kg
15) Fuel Management	Core Reloading Batch-20%
16) Average Fuel Burnup	
Standard Rods	18,700 ^{MWD} MTU
Corner Rods	20,900 ^{MWD} MTU
Core Average	18,980 ^{MWD} MTU
17) Peak Average Burnup Ratio	1.84

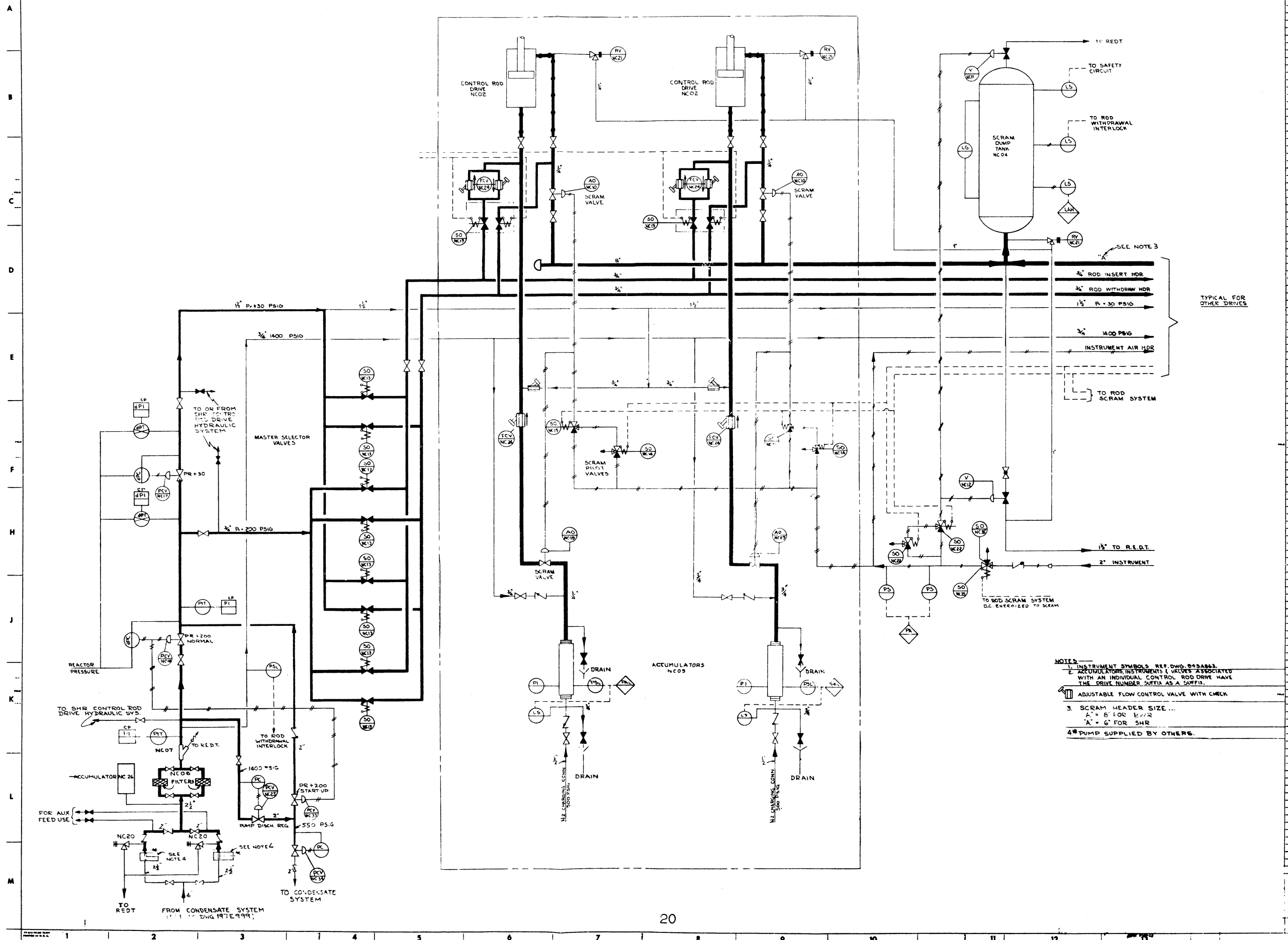
NOZZLE SCHEDULE			
NO	QTY	FUNCTION	LOCATION
1	3	COOLING WATER INLET	BOTTOM HEAD
2	3	COOLING WATER OUTLET	VESSEL BODY
3	3	RUPTURE DETECTION	VESSEL BODY
4	4	INSTRUMENT	VESSEL BODY 2 TOP HEAD 2
5	1	LIQUID POISON & CORE SPRAY	VESSEL BODY
6	2	FEEDWATER SPARGERS	VESSEL BODY
7	4	STEAM OUTLET	VESSEL BODY
8	10	PRSS. RELIEF	TOP HEAD
9	1	DRAIN	BOTTOM HEAD
10	25	IN CORE FLUX MONITOR	BOTTOM HEAD
11	69	CONTROL RODS	BOTTOM HEAD
12	1	VENT	TOP HEAD
13	1	DRYBOX DRAIN	VESSEL BODY



UNLESS OTHERWISE SPECIFIED USE THE FOLLOWING:			
APPLIES PRACTICES	SURFACES	FINISHES	UNIT
✓	✓	✓	✓

WILE ASSEMBLY
CONTROL ROD DRIVE
PRINT MADE FOR MCAP 1207





- NOTES:**
1. INSTRUMENT SYMBOLS REF. DWG. 043A863.
 2. ACCUMULATORS, INSTRUMENTS (VALVES ASSOCIATED WITH AN INDIVIDUAL CONTROL ROD DRIVE HAVE THE DRIVE NUMBER SUFFIX AS A SUFFIX).
 3. SCRAM HEADER SIZE ...
A" = 6" FOR E/R
A" = 6" FOR SHR
 4. PUMP SUPPLIED BY OTHERS.

2.2 Superheat Reactor

The superheat reactor is a single pass, steam cooled and light water moderated reactor. Saturated steam from the boiling water reactor enters nozzles at the top of the vessel and flows through demisters and inlet tubes down into the core. The steam is superheated as it passes downward through the core and is collected in an insulated manifold system at the bottom of the vessel. Superheated steam leaves the vessel through nozzles in the side of the vessel below the core.

Boiling of the moderator occurs due to heat generation in the moderator and to heat losses from the superheated portion of the core and manifold system. The moderator steam formed flows upward to the water surface where it joins the saturated steam coming from the boiling water reactor. Feed water required to maintain moderator level in the vessel is introduced through a nozzle in the side of the vessel near the bottom.

The superheat reactor shown on the isometric drawing on the following page is designed for these conditions:

Total thermal power	213.8 MW
Inlet steam pressure	1055 psia
Exit steam temperature	900°F
Fuel clad	UO ₂ -S.S. 304
Fuel power density	20 KW/kg of U
Core power density	36.6 KW/liter

A complete superheat reactor data sheet is included at the end of Section 2.

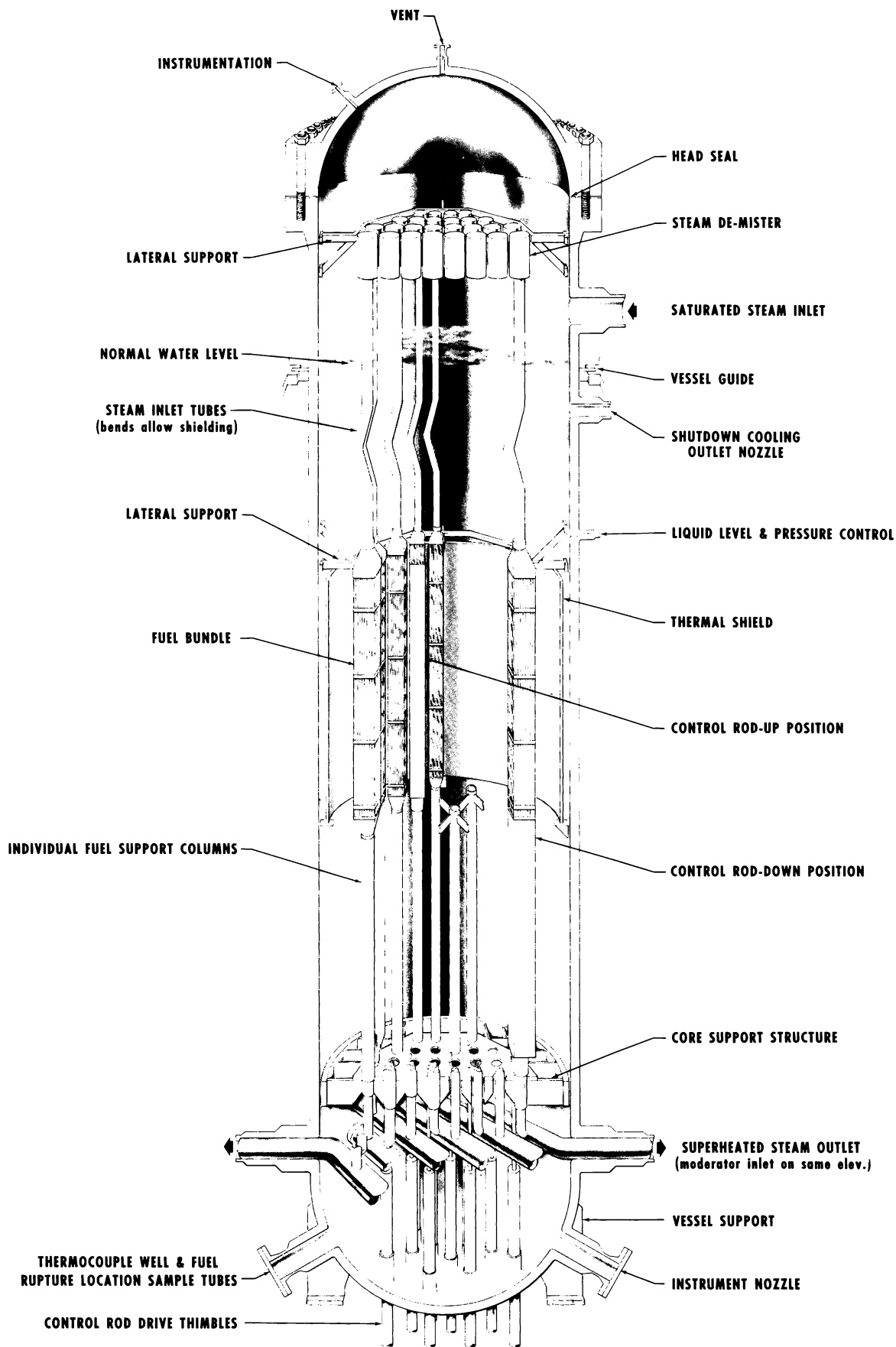
2.2.2 Reactor Vessel

The superheat reactor vessel is designed to conform with the same criteria and codes as the boiling water reactor vessel (Section 2.1.2).

2.2.3 Vessel Internals

The core is supported by a stainless steel grid structure mounted on lugs at the bottom of the vessel. Fastened to this structure are separate support columns for each fuel assembly. Approximately half of the columns in this structure serve the additional purpose of guiding the lower end of the control rods. The structure is stiffened with gussets to minimize deflections and internal lateral supports are provided.

Fuel assemblies are coupled to the support columns which also serve as steam conduits. The fuel elements are an assembly of insulated rods in a square array. Each rod consists of an annular UO₂ Fuel region clad both inside and outside with stainless steel. The fuel tube is inserted in a process tube of zircaloy lined with stainless steel. Spacing and support of the rods is provided by a series of grids along the element. These grids also contain rollers to guide the control elements.



SINGLE PASS SUPERHEAT REACTOR

The control rod is a hollow square tube which encases alternate fuel assemblies. The absorber section is fabricated of boron stainless steel. The control elements are guided during insertion and withdrawal by rollers coupled to bottom mounted drive mechanisms.

Inlet tubes are connected to the top of each fuel assembly and extend upward to the demisters near the top of the vessel. These tubes displace the water shielding above the core and are designed with offset bends to prevent radiation streaming.

Accumulated moisture in the saturated steam is removed by demisters at the top of the steam inlet tubes. The demisters are fastened to and supported by the inlet tubes.

Lateral supports are located at the top of the fuel assemblies and at the demisters.

The steam leaving the fuel assemblies is discharged to a collecting header through the hollow fuel support columns previously described. The header assembly, located between the core support structure and the control rod drive thimbles, is shop fabricated to assure adequate quality control and to simplify field assembly.

2.2.4 Control Drives

Control drives for this reactor are essentially identical to those provided for the boiler (Section 2.1.4).

The control rod is in the form of a 8.65 in. square tube. This control rod shape has excellent mechanical strength when compared to a cruciform rod of the same blade thickness. The small blade thickness results in minimum water-gap flux peaking.

The control rod is enclosed over a precision upright pipe column and attached to the offset control rod drive. The fuel assembly is located and positioned on top of the pipe column. Located at various positions on the fuel assembly are control rod rollers which maintain the position of the control rods with respect to the fuel assembly.

2.2.5 Core Design

2.2.5.1 Core Design Criteria

The superheat reactor core design was based on the following criteria;

- (1) The maximum clad temperature for 100 percent power in the hot channel is to be 1250°F
- (2) The maximum fuel temperature under transient overpower conditions is to be less than 4500°F.
- (3) The stainless steel cladding has to be of sufficient strength to maintain fission gas.

- (4) The reactor should be designed for minimum reactivity change for "flooding" or "unflooding" the steam coolant channels.
- (5) Within the limitation of criteria (1) through (4), the reactor should be designed for low power costs.

2.2.5.2 Single Pass Selection

Detailed analyses were performed on several fuel elements with differing arrangement, geometry, and operating conditions. The single pass was selected as the reference design because of the following advantages over double pass systems considered:

- (1) It allows a higher power density core. The single pass coupled with hollow cylindrical fuel allows the greatest volumetric heat generation within the clad temperature limit of all concepts considered.
- (2) It reduces the flow area required to maintain a given pressure drop. Reduction of flow area minimizes the reactivity changes associated with flooding or voiding normal steam passages.
- (3) It reduces the difference between average temperature of the inner and outer clad and thus minimizes differential expansion and resulting stresses in the fuel element.
- (4) It permits an increase in steam outlet temperature for a given clad temperature limit. In the second pass of any two pass system, the steam coolant must pass through the peak heat generation area at a high temperature. This reduction in heat absorbing capability requires a reduction in outlet steam temperature in order to maintain the limiting clad temperature.
- (5) It reduces mechanical design problems and operational requirements. Single pass designs eliminate the need for separating inlet and outlet steam and allow for draining of the superheater fuel prior to startup.

2.2.5.3 Superheat Fuel Assembly

The superheat fuel assembly consists of an inlet steam line with de-mister, and a fuel bundle containing 49 fuel elements. The thermally insulated lower end of the fuel bundle contains a gasketed joint which seals a seating surface provided by an upright pipe column. The force available for sealing is made up of the load of the fuel assembly, the pressure of the water acting on the tube sheet and the pressure differential between the superheater steam pressure and the saturated steam pressure.

Superheat Fuel Element

The basic superheat fuel channel consists of a zircalloy process tube and a stainless steel insulation liner around an annular fuel rod. The fuel rod is clad internally and externally with stainless steel. An annular gap between the process tube and the insulation liner contains stagnant steam to serve as a thermal insulation between the high temperature superheated steam and the moderator water. This thermal insulation permits the zircalloy process tube to operate at moderator temperature. The fuel element process tube has the function of withstanding the pressure differential between the moderator and the superheated steam and also provides the structural support between the upper and lower tube sheets. The process tube is designed to withstand 300 psi pressure differential between the moderator and the steam. The stainless steel liner has the function of defining the coolant flow passage as well as acting as the insulating barrier between the fuel and the process tube. This liner holds the fuel rod centered in the process tube by means of radial embossed surfaces on the internal and external surfaces of the liner. The liner is retained at the inlet end of the fuel element but is free to expand at the lower end which will allow relative thermal expansion with respect to the fuel rod and the process tube.

The fuel rod is supported from a plate at the upper end of the fuel assembly. The rods hang freely from this support plate to accommodate relative thermal expansion with respect to the liner.

2.2.5.4 Heat Transfer and Hot Channel

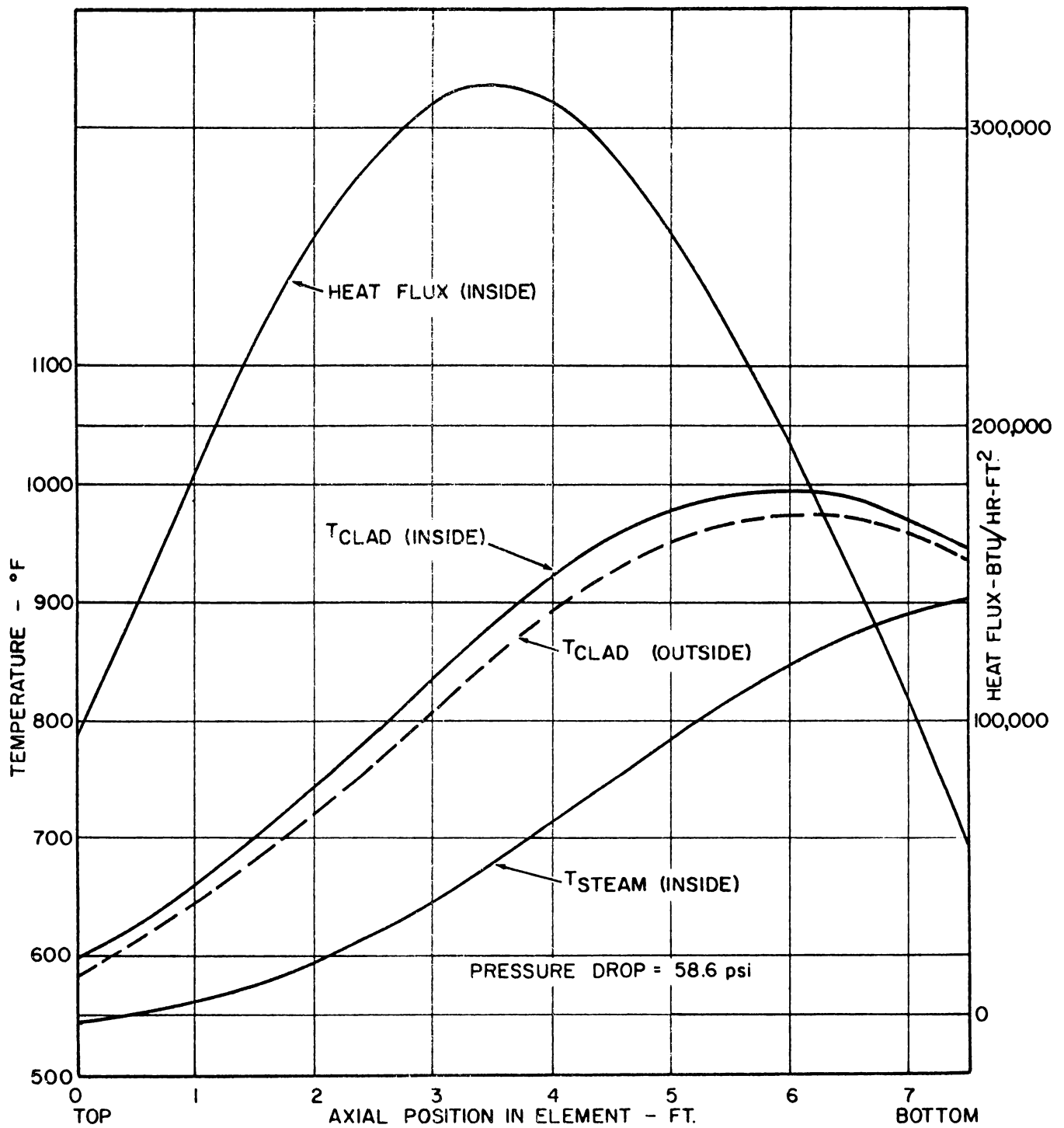
The superheat core is orificed to obtain approximately the same steam temperature from all 52 fuel assemblies. Permanent orifices are used in the steam exit lines to accomplish most of the orificing. The centrally located fuel bundles are unorificed. Additional orificing may be used in the steam inlet lines to permit orificing adjustment at the time of core refueling and fuel relocation.

Central Channel

A central fuel channel when delivering 900°F steam operates at an exceptionally low clad temperature of 992°F. The heat transfer characteristics of this central fuel channel were computed with the aid of the digital SPAM Code* and are shown in graphical form on the following page. The important results are:

- | | |
|------------------------------|-------|
| (1) Exit steam temperature | 900°F |
| (2) Maximum clad temperature | 992°F |

* Steady state analysis of single pass annular superheat fuel elements.



FUEL ELEMENT IN CENTRAL CORE REGION

(3) Average heat flux	175,000 Btu/hr-ft ²
(4) Maximum heat flux	312,000 Btu/hr-ft ²
(5) Pressure drop	58 psi
(6) Maximum fuel temperature	2420°F

Hot Channel Characteristics of Steam Cooled Reactor

A steam cooled reactor has two undesirable characteristics. First, the specific heat of superheated steam decreases as temperature decreases. Second, the steam flow in a hot channel will decrease due to the increased pressure drop of the hotter steam. The following tabulation shows the effect of a 20% increase in local fuel element power of a central channel.

	<u>Central Channel at 100% Power</u>	<u>Central Channel at 120% Power</u>
Pressure Drop	58 psi	58 psi
Steam Flow	1280#/hr	1216#/hr
Exit Steam Temperature	900°F	1021°F
Steam Temperature Rise	355°F (100%)	476°F (134%)
Maximum Clad Temperature	991	1156

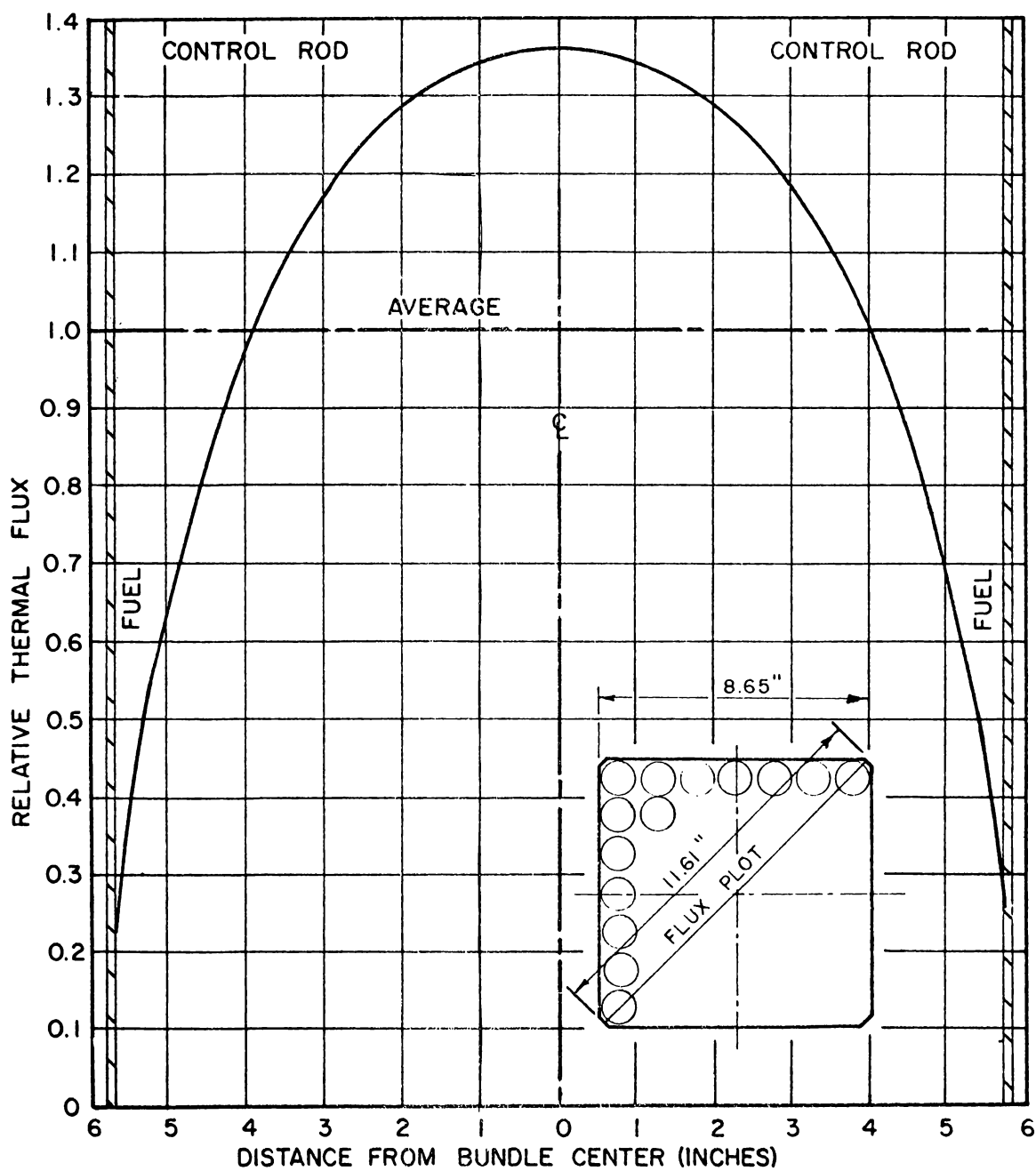
It should be noted that only an increase in local power of 20%, without considering any other hot channel factor, increase the steam temperature rise by 34% and increase the clad temperature by 165°F.

Local Peaking Effects Due to Control Rod Position

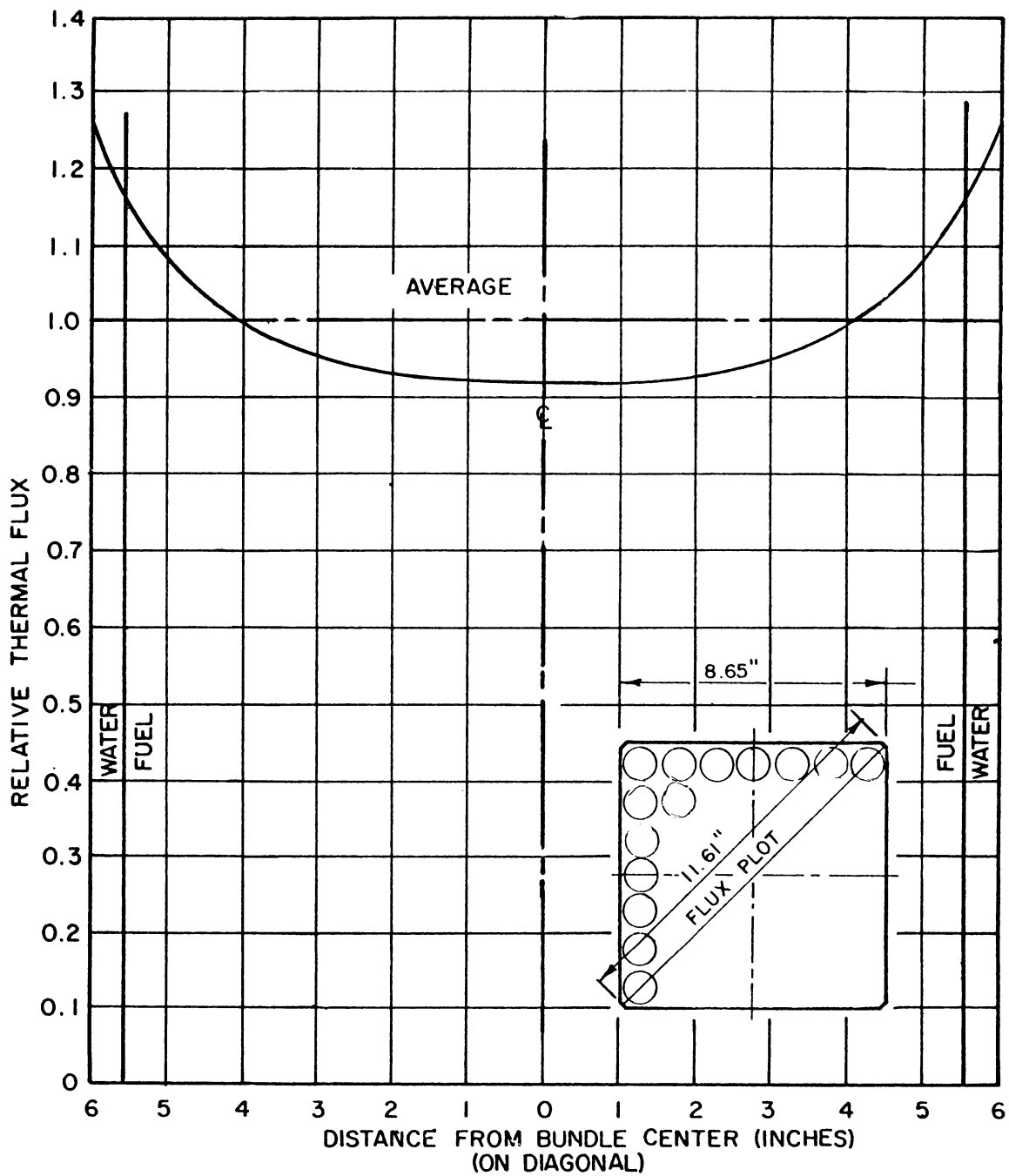
The movement of control rods in and out the reactor will change the local distribution. The effects of control rods position is shown on the two following figures. An additional local flux peaking effect will result if adjacent control rods are not inserted into the reactor to the same depth. In order to obtain the satisfactory axial power distribution throughout core life, it will be necessary to withdraw adjacent control rods different distances.

The local peaking effects of these three control rod patterns are:

	<u>Local Power</u>	<u>Local Heat Flux</u>	<u>Hot Rod</u>
Control rod fully in	1.30	1.30	Center
Control rod fully out	1.10	1.25	Corner
Alternate rods out	1.35	1.35	Center



RELATIVE THERMAL FLUX PLOT ACROSS
ONE FUEL ASSEMBLY-RODS IN



**RELATIVE THERMAL FLUX PLOT ACROSS
ONE FUEL ASSEMBLY-RODS OUT**

Hot Channel Clad Temperature

The most important consideration in determining the hot channel clad temperature is the local power shape. The reactor of this study has been designed to operate at full power with alternate control rods inserted 80%, or with a local channel power of 120% and a local heat flux of 135%. The reactor has been designed on the basis that this local peaking will occur, not on the basis that it may occur. As this one condition (local power) is so important, probability type calculations have not been used on the hot channel analysis.

The hot channel is assumed to have these conditions:

- (1) Hot channel is in the central region of the core.
- (2) 120% channel power.
- (3) 135% local heat flux peaking.
- (4) 10% reduction in heat transfer condition due to fowling of the heat transfer surfaces.
- (5) 2 mil dimensional variation or cladding or process tube.

The hot channel, shown graphically on the following pages, has these characteristics:

- | | |
|------------------------------|--------------------------------|
| (1) Pressure drop | 58 psi |
| (2) Exit steam temperature | 1 044°F |
| (3) Maximum heat flux | 420,000 Btu/hr-ft ² |
| (4) Maximum clad temperature | 1233°F |

2.2.5.5 Safety Considerations

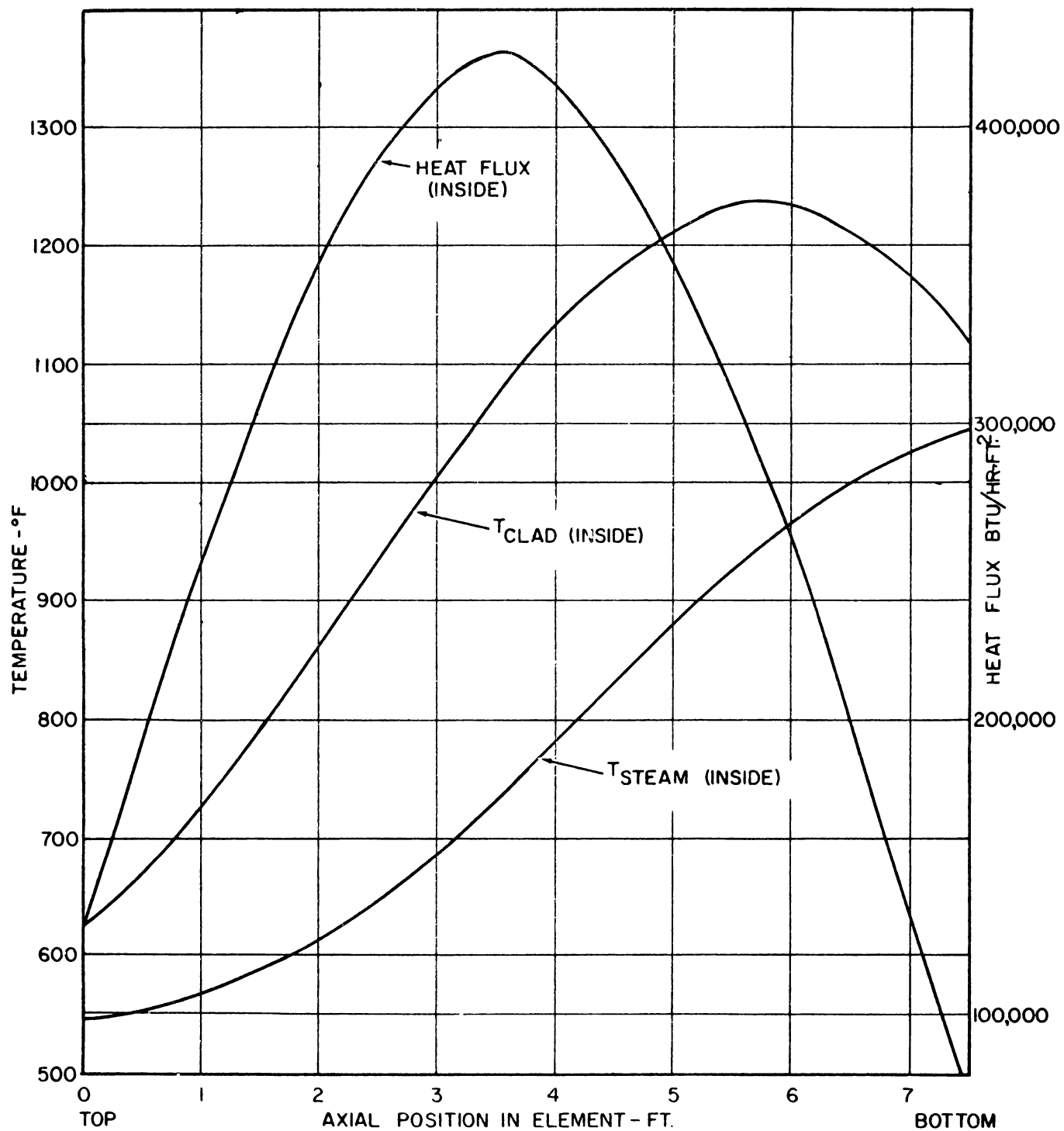
Four accidents which are of importance in superheat reactor design are:

- (1) Flooding of the steam coolant passages when the reactor is operating at power.
- (2) Steam pipe rupture accidents.
- (3) Sudden loss of steam flow.
- (4) Cold start-up accidents.

The design features incorporated in this superheat reactor for these four accidents are discussed in the following paragraphs:

Hot Flooding Accident

- (1) The hot flooding accident, is a nuclear excursion produced by the sudden addition of water moderator into the core volume occupied by the steam coolant. The change in water to fuel ratio may result in a positive reactivity addition. This problem has been attacked by both making the accident very improbable and also by reducing the severity of the accident.



HOT CHANNEL FUEL ELEMENT

Steps taken to make the accident very improbable are:

- (a) The reactor design has been changed to a single pass design with steam exits in the lower portions of the reactor vessel. A hot flooding accident now has to be based on the premise that the exit steam lines are suddenly plugged up and water is suddenly introduced into the core:
- (b) As will be discussed under a pipe rupture accident, the process tubes and steam internals will not collapse.
- (c) The full power rate of feedwater and rod seal water addition to the superheat reactor vessel is 66,000 #/hr, or equivalent to a water change of less than 0.1 inch per second. Similarly the water introduction rate (66,000 #/hr) is so low that the 80 seconds would be required to inject sufficient water to occupy the core steam volume. Therefore, fast flooding due to level control failure is incredible.

Steps taken to reduce the severity of the accident are:

- (d) The use of a single pass design reduces the steam coolant volume. The steam coolant occupies only 14% of the core volume.
- (e) Burnable poisons have been incorporated into the fuel to reduce the amount of excess reactivity that is present in the core at any time.
- (f) The moderator to fuel ratio has been selected on the basis of minimizing reactivity changes due to flooding and unflooding. The selected moderator to fuel ratio of 2.6 is above the moderator to fuel ratio for minimum power costs.

Due to the steps of (d), (e), and (f), the estimated maximum increase in reactivity under the worst combination of core exposure, core flooding, and control positions is 2-3% reactivity. Analog computer studies of flooding accident with reactivity increasing at the rate of 4 dollars/seconds show that the maximum hot spot clad temperature increase would be 150-200°F.

In summary, complete and sudden flooding of the core is very improbable and the accident is not severe.

(2) Pipe Rupture Accident

Rupture of the steam entrance or exit lines is of concern due to the fact that pipe rupture is a possible method of core flooding and a possible method of stopping steam flow.

The rupture of an exit steam pipe will increase the reactor steam velocity and will also increase the core pressure drop. The internal steam piping has been designed for 500 psi pressure differential and the core process tubes have been designed for 300 psi differential in order to prevent process tube or pipe collapse.

The pipe connection between the boiling water reactor and the superheat reactor has been equipped with check valves. The check valves, located at the inlet nozzles to the superheat reactor, prevent the loss of superheat moderator and steam in event of a pipe rupture between the two reactor vessels.

(3) Sudden Loss of Steam Flow

The superheat reactor has been designed for complete and instantaneous loss of coolant steam. The emergency cooling system, described in Section 4.9, will limit the maximum clad temperature rise to approximately 200°F. The transient temperature performance were investigated by using the digital TIGER Code.

(4) Cold Start-up Accidents

The cold unflooding reactivity and the cold temperature coefficients will be negative through most of the core life. However, at the end of life of the core the cold unflooded reactivity coefficient may be 0.5% positive. To prevent the necessity of start-up with a cold moderator, the superheat moderator is heated by interchanging moderator between the boiling water reactor and the superheat reactor. The superheat reactor is always sub-critical until the moderator is heated and steam flow through the superheat reactor is established.

2.2.5.6 Physics Calculations

Superheat Reactor Physics Qualifications

1. The superheat reactor of this study differs from previously investigated superheat reactors in that a burnable poison is used. A burnable poison has been used to reduce the maximum excess reactivity. A smaller excess reactivity is important in reducing the severity of "flooding" and "unflooding" accident.
2. Most of the physics calculations performed on this study were concerned with the change in reactivity with flooding, unflooding, moderator void and temperature coefficients, and with the determination of local flux distribution with control rods in, control rods out, and with alternate control rods out.
3. No detailed burnup studies were made. The initial k_{eff} required to reach 19,800 MWD/MTU of U average burnup for the hot clean core without burnable poison was estimated from previous point burnup calculations. The calculations, while including equilibrium fission product poisoning, were on a single batch basis and did not take into account any spatial variations in burnup. While it is known that under a multiple batch reloading scheme a longer irradiation of the fuel can be achieved for the same enrichment than in the single batch case, it is also known that the axial variation in burnup will tend to cancel this gain for the first core. When the equilibrium core is reached, it is expected that approximately a 20% gain in discharge exposure can be attained over first core discharge. However, the burnable poison penalty (due to incomplete poison burnup) was calculated on the basis of discharge exposure concentration with no allowance for spatial distributions which underestimates this penalty. This will tend to cancel the multiple batch reloading gains for the equilibrium core. It is therefore felt that the required initial enrichment that has been estimated to reach 19,800 MWD/MTU of U average discharge exposure for the equilibrium core is realistic. The Uranium-235 depletion and Plutonium buildup were extrapolated from point burnup studies undertaken for boiling water reactors⁽¹⁾ and represent discharge isotopic concentrations. These estimates are considered valid as they were interpolated for the actual surface/mass ratio, initial enrichment, and water to fuel ratio.

For a water to fuel ratio of 2.6; or stainless to fuel ratio of 0.14, and a steam void volume of 0.14, the burnup has been estimated as follows:

(1) Equilibrium discharge exposure	19,800 MWD/MTU
(2) Initial enrichment	3.5 a%
(3) Final enrichment	1.65 a%
(4) Equilibrium plutonium discharge enrichment	0.7 a%

(1) R. L. Crowther and W. L. Cranor, Uranium-235 Depletion and Plutonium Buildup in Large Boiling Water Reactors, April 17, 1959, GEAP-3151.

SUPERHEAT REACTOR DATA SHEET

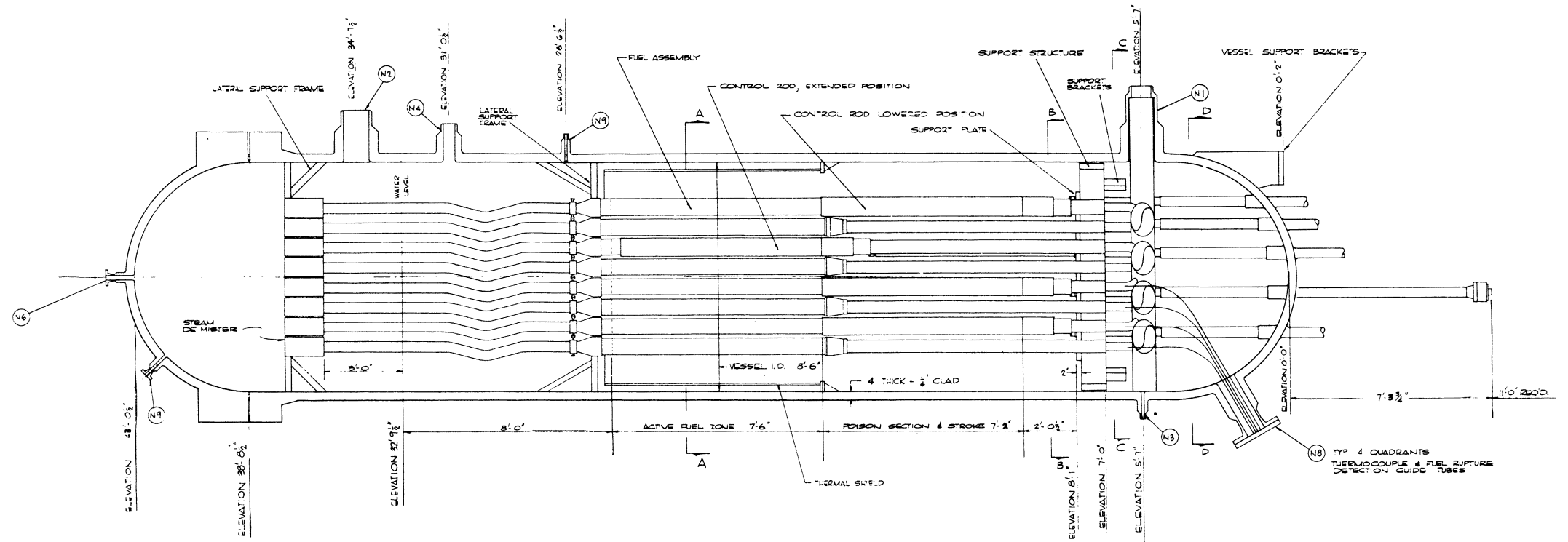
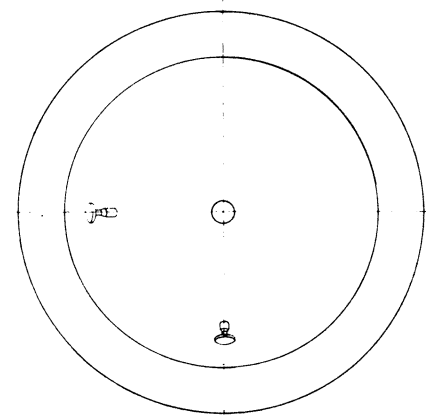
2.2.6

A. Thermal Power	
1) Thermal Power (to steam)	198.8 MWT
2) Thermal Power (to moderator)	15.0 MWT
3) Total Thermal Power	213.8 MWT
B. Steam Conditions	
1) Exit Steam Condition	980 psia, 900°F
2) Inlet Steam Condition	1,055 psia
3) BWR Steam Flow	2,564,000 lbs/hr
4) SHR Moderator Steam Flow	52,000 lbs/hr
5) Total Steam Flow	2,616,000 lbs/hr
C. Reactor Description	
1) Reactor Vessel	
a. Inside Diameter	8.50 ft.
b. Inside Height	43.00 ft.
c. Wall Thickness	4 in. base 1/4 in. S.S. clad
d. Material - base metal	ASTM-A-302 Grade B
wall clad	ASTM-A-264 Grade B
e. Design Pressure	S.S. type 304 - modified 1310 psig
f. Design Temperature	650°F
2) Reactor Core	
a. Active Equivalent Diameter	5.83 ft.
b. Active Height, ft.	7.5 ft.
c. Active Core Volume	201.0 ft ³
d. Total Uranium loading	10,720 kg
e. Average Uranium Content	2.57 a%
Initial Uranium Content	3.50 a%
Final Uranium Content	1.65 a%
Plutonium at end of life	0.70 a%
f. Structural Material	304 S.S.
g. Neutron Moderator	light water
h. Moderator to Fuel Ratio	2.60
3) Reflector	
a. Material	light water
b. Axial Thickness	8 ft.
c. Radial Thickness on equivalent diameter	13 in.
4) Fuel Elements	
a. Fuel Material	UO ₂
b. Fuel Element Geometry	Annular, 0.260 I.D., 0.692 O.D.
c. Clad Material	304 S.S.
d. Fuel Meat Thickness	0.216 in.
e. Clad Thickness, in.	0.012
f. Fuel Form	Resonant Compacted
g. Gap Filler Material	Helium
5) Fuel Assemblies	
a. Total Number	52
b. Number of Elements per Assembly	49
c. Cross Section Assembly	8.65 in.
d. Process Tube Material	Zircaloy-2
e. Process Tube Dimension	0.932 O.D. x 0.025 wall
f. Process Tube Insulation	0.008 304 SS Liner and Stagnant Steam
g. Process Tube Spacing	1.18 in.
h. End Fitting Material	Zircaloy and SS 304

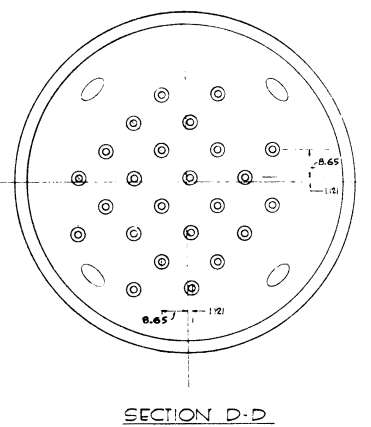
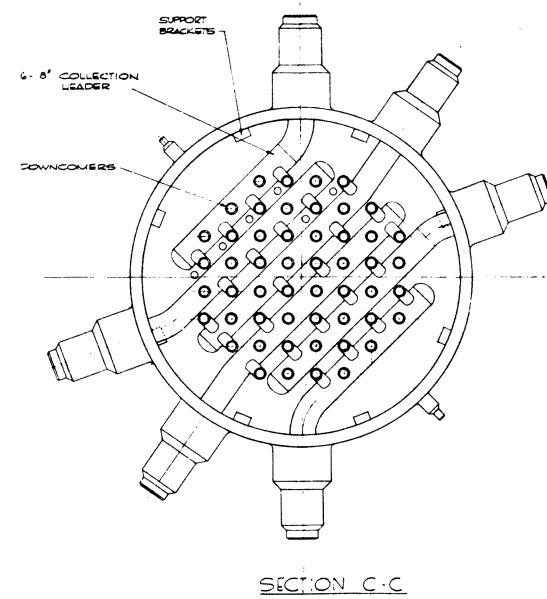
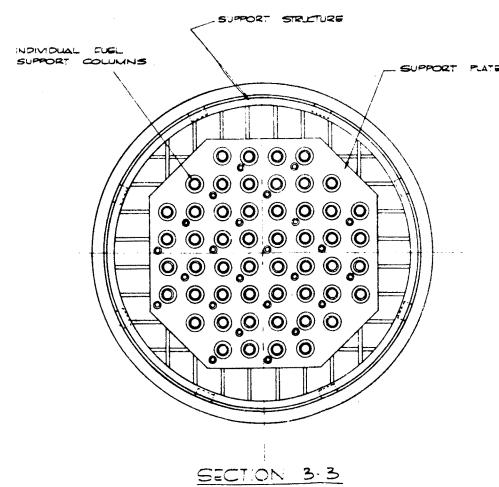
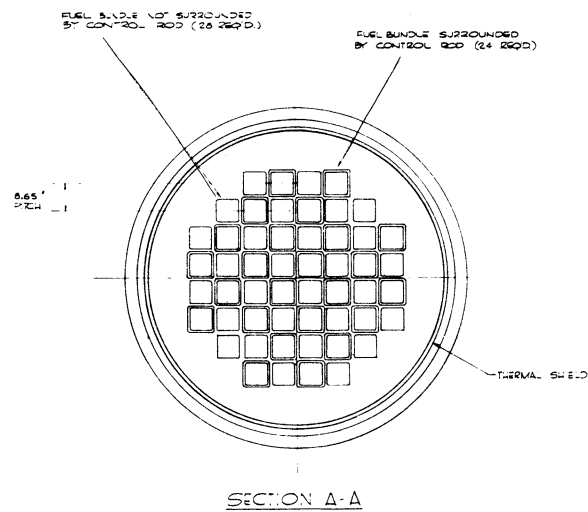
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|----|----------------------------|-----------------------------|
| 6) | Reactor Control | |
| a. | Method of Control | Control Rod Movement |
| b. | Absorber Material | Boron Steel |
| c. | Number of Control Elements | 24 |
| d. | Cross Sectional Dimensions | 8.65 x 8.65 in. square |
| e. | Effective Length | 7.17 ft. |
| f. | Type of Drive | Hydraulic Locking
Piston |

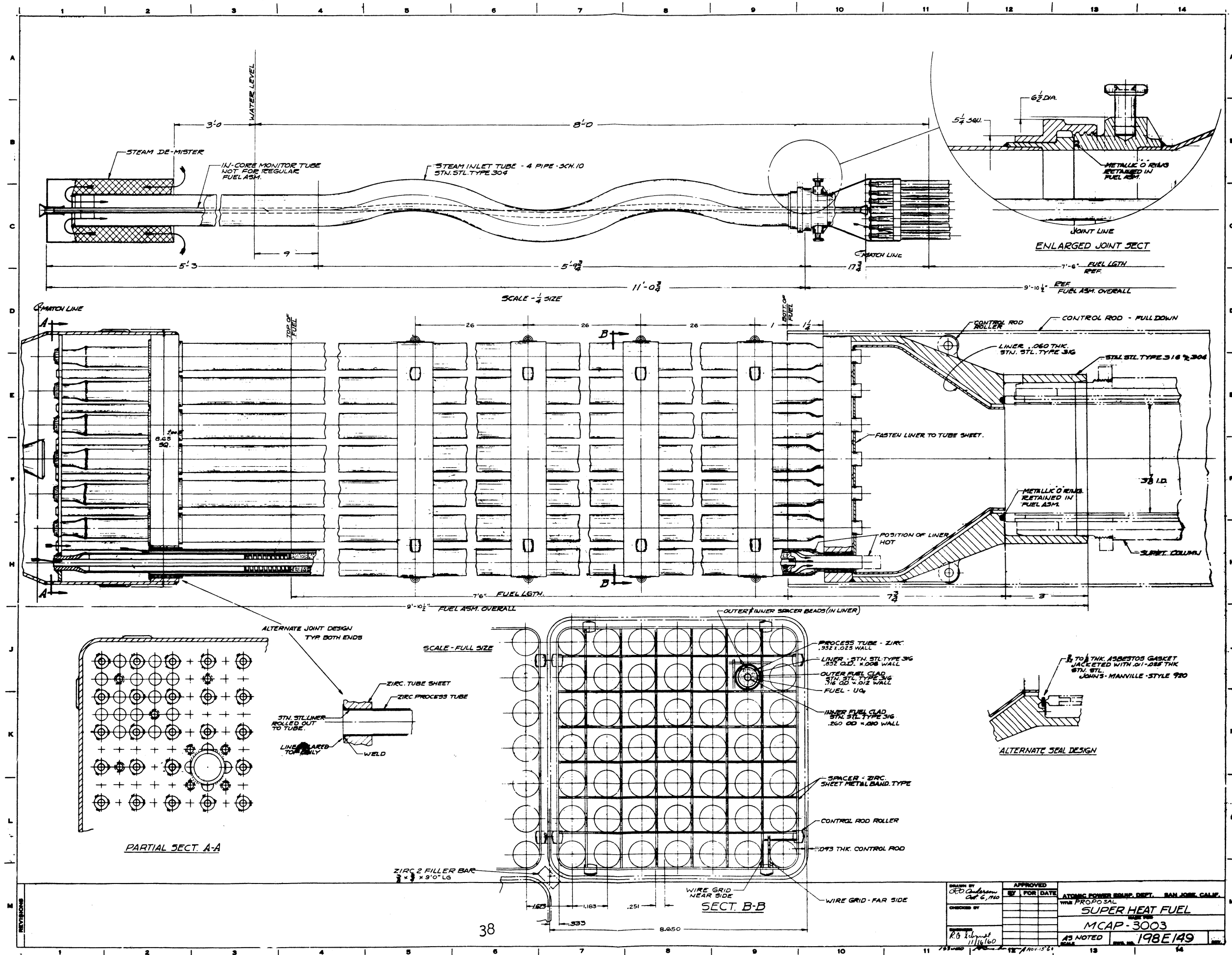
D. Performance Data

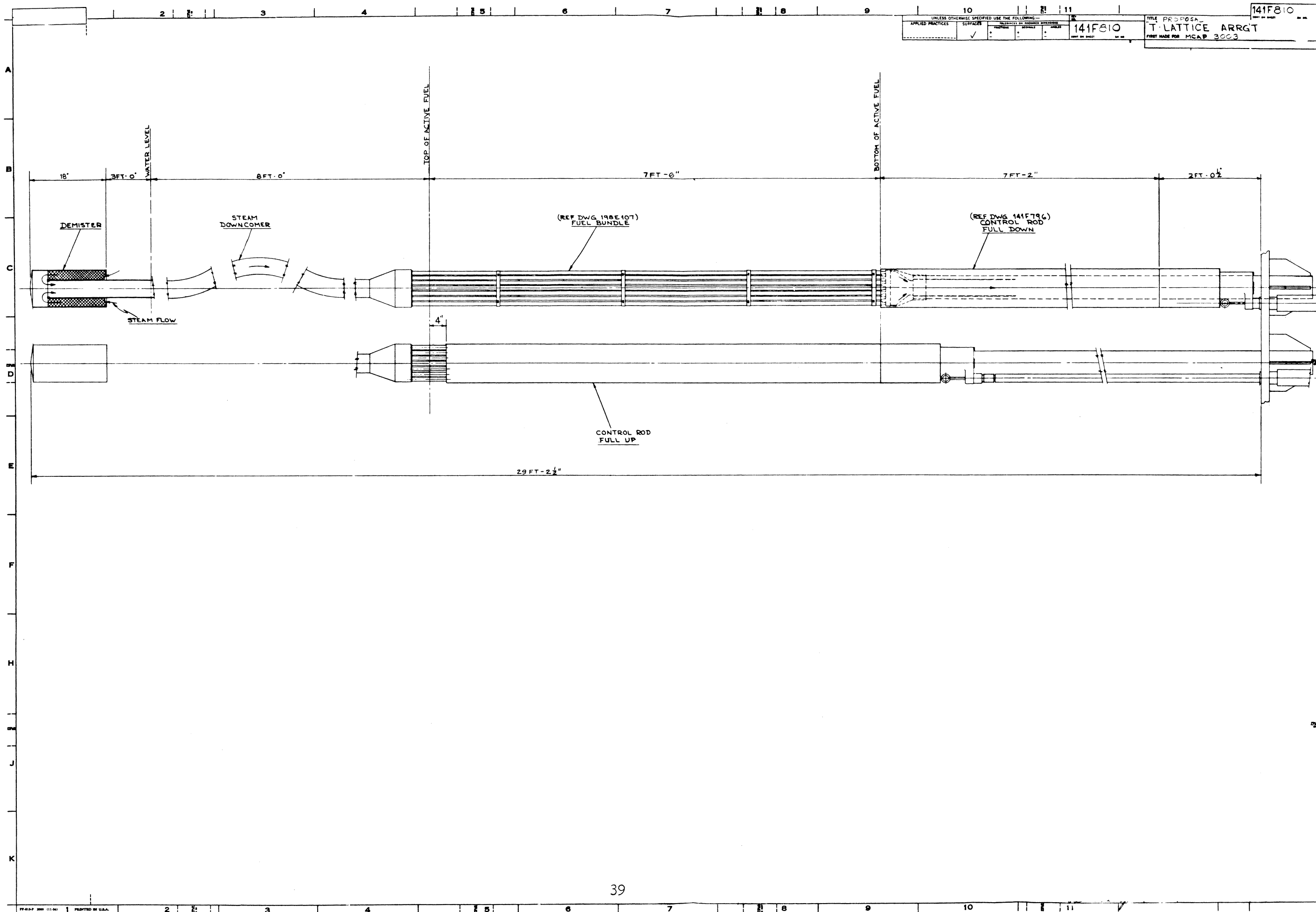
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|-----|--|---|
| 1) | Reactor Coolant | Steam |
| 2) | Reactor Coolant Outlet Temperature | 900°F |
| 3) | Reactor Coolant Inlet Temperature | 553.5°F |
| 4) | Steam Pressure | 1055 psia inlet
980 psia outlet |
| 5) | Steam Coolant Flow | 2,616,000 lbs/hr |
| 6) | Average Core Coolant Flow | 89.5 ft/sec-entrance
167 ft/sec-exit |
| 7) | Max. Fuel Temperature (125% Power) | 3800°F |
| 8) | Max. Clad Temperature (100% Power) | 1250 |
| 9) | Max. Core Heat Flux (125% Power) | 525,000 Btu/hr-ft ² |
| 10) | Average Core Heat Flux | 140,000 Btu/hr-ft ² |
| 11) | Average Core Power Density | 1,070 KW/Ft ³ |
| 12) | Peak to Average Power Ration | |
| | Axial | 1.40 |
| | Radial | 1.25 |
| | Local | 1.35 |
| | Overpower | 1.25 |
| | Inside clad surface/average clad surface | 1.26 |
| | Total | 3.70 |
| 13) | Average Specific Power | 20 KW/Kg |
| 14) | Fuel Management | Fuel Batch-20% |
| 15) | Average Fuel Burnup | 19,800 MWD/MTU |
| 16) | Peak/average Burn-up Ratio | 1.77 |

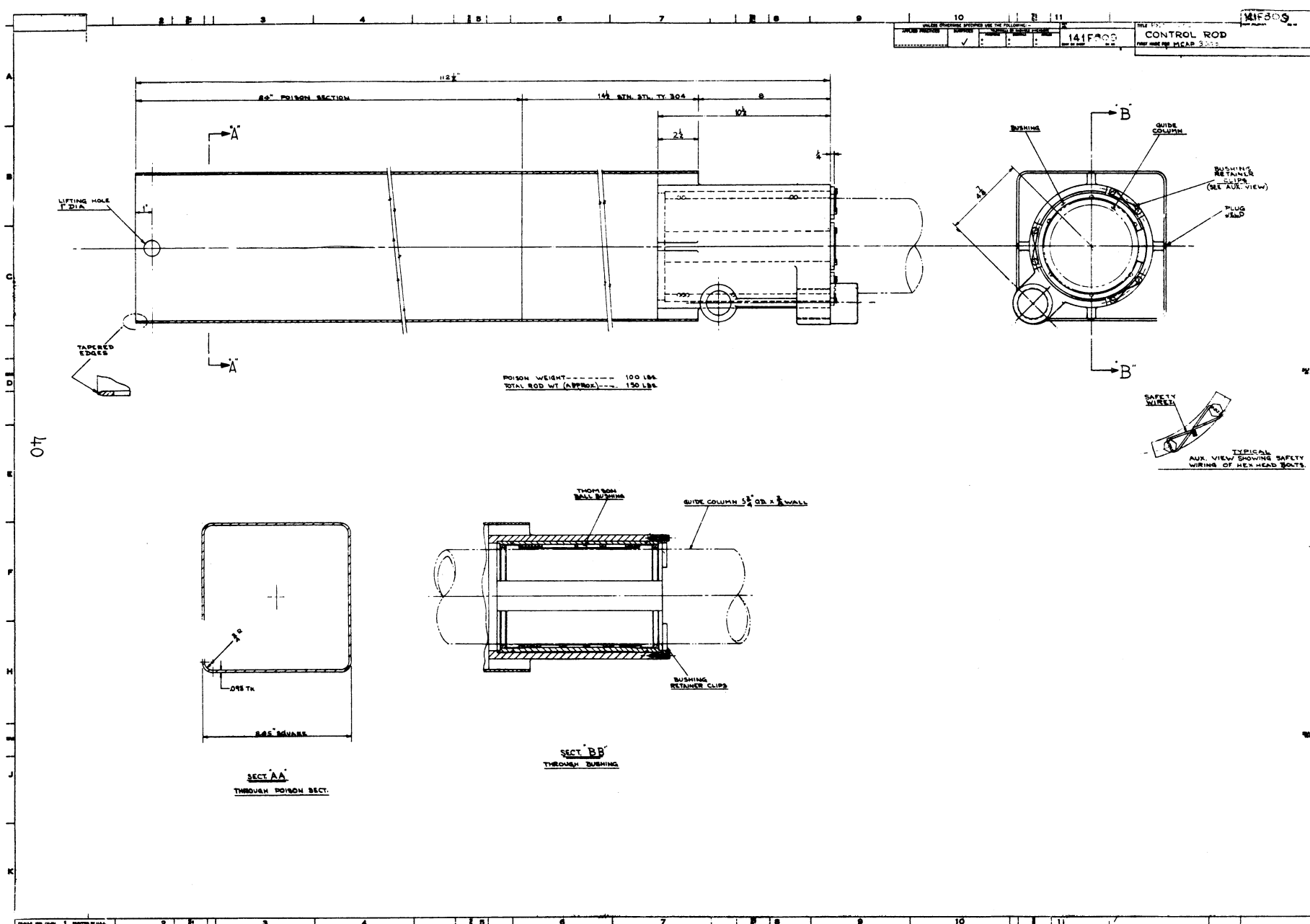


NOZZLE SCHEDULE			
NO.	DESCRIPTION	SIZE	LOCATION
1	4 SUPHEAT OUTLET	0" THERMAL SHIELD	ELEV. 5'7"
2	4 HOT STEAM INLET	0" THERMAL SHIELD	ELEV. 35'7 1/2"
3	2 THERMAL SHIELD	2 1/2" THERMAL SHIELD	ELEV. 5'7"
4	24 CONTROL ROD DRIVE	6"	ELEV. 32'0 1/2"
5	1 VESSEL VENT	1 1/2"	TOP HEAD
6	1 VESSEL DRAIN	1 1/2"	BOTTOM HEAD
7	4 INSTRUMENT	0"	BOTTOM HEAD
8	4 LIQUID LEVEL & PRESSURE GAGE	1"	ELEV. 27'6 1/2" 2 TOP HEAD









3.0 Reactor Building

The reactor building provides shielding and containment for the boiling water reactor, the superheat reactor, the reactor auxiliaries, and fuel storage facilities.

The design and arrangement of this building is shown on drawing 198E124, SH. 1 and 2. These drawings are located at the back of this section.

3.1 Basis of Design

The design of the entire plant is based on the premises that:

- a) Normal operation of the plant must not result in the exposure of any persons on or off plant premises to radiation in excess of the current recommendations of recognized national and international radiation protection groups.
- b) Safety against a nuclear accident that might release dangerous amounts of radioactive materials must be preserved even in the event of equipment malfunction, operator errors, or other credible contingencies.

From the structural standpoint, two basic requirements are governing:

- a) Shielding must be provided around radioactive equipment to comply with (a) above.
- b) A reactor enclosure adequate to confine any significant quantities of radioactive materials released from a serious credible accident must be provided to comply with (b).

3.2 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/s/hr</u>
Substantial to continuous	1
Less than 10 hours per week	6
Less than 5 hours per week	12

3.3 Reactor Building Shielding

A suggested design of the reactor enclosure is included in this proposal. Generally, the containment areas are accessible with suitable provision for access control. Operating floors, corridors,

and certain rooms are accessible at all times. These areas have been shielded to reduce radiation levels to less than 6 mrem/hr. Certain areas are not accessible during operation. These are:

- Reactor dry wells
- Primary steam pipeway
- Reactor cleanup demineralizer cell

The design of the dry wells and top shielding is based on full accessibility to the area above the reactors during operation. Dry well shield covers are selected to reduce radiation intensity on the service floor to 1 mrem/hr. During refueling the area in the dry well above the reactor vessel will be flooded. The depth of water is sufficient to maintain the desired 1 mrem/hr.

3.4

Reactor Enclosure

The confinement requirement is provided by a pressure suppression type reactor enclosure. In the pressure suppression system the reactors are installed in pressure-tight dry wells. These dry wells are vented to a water-filled suppression pool by a series of vent pipes. In the event of an incident where primary fluid is released from the reactor system the energy is released as steam to the dry well. Following an initial pressure rise in the dry well, the steam vents to the suppression pool where it is condensed. Experimental programs conducted by General Electric have conclusively demonstrated that all of the steam released will be condensed within the suppression pool. Certain non-condensable gaseous fission products will also be released with the steam. These are collected in the area above the suppression pool - a leak-tight concrete structure. No significant pressure rise is encountered in the suppression pool.

The complete isolation of the pressure suppression system permits conventional design and construction techniques to be applied to the remainder of the reactor building. The principal exceptions to this are that:

1. certain portions of the structure must be designed to meet shielding requirements,
2. the structure must be reasonably leak-resistant in order to protect the small connections from the primary system which will be brought into the general building areas for control and instrumentation purposes,
3. the refueling building must be leak-resistant in order to provide protection against fission product leakage in the remote event of a refueling accident.

For the proposed plant, the pressure suppression system design is based on a postulated maximum credible accident involving release of all the hot water energy in both reactor vessels at a rate consistent with that which would result from two openings the size of the saturated steam line connection between the boiler and superheat reactor vessels. The dry well design pressure is 100 psig.

The dry well vessel, vent header, and connecting pipes will be designed, constructed, and tested in accordance with Section VIII of the ASME Boiler and Pressure Vessel Code and related case decisions which deal specifically with the application of the Code to containment vessels for nuclear reactor systems.

The pressure suppression pool is a reinforced concrete structure located on the north side of the reactor building. The pool is completely enclosed by reinforced concrete walls and a roof slab with radiation shielding protection provided by the walls, where required. The design pressure for the space above the suppression pool is 8 psig. The reactor building is a reinforced concrete structure that houses the reactor vessel and related equipment which comprise the nuclear steam supply system, including the fuel storage pool and fuel handling facilities.

Where shielding requirements permit, the exterior walls of the building are concrete block. The operating floor, which is the top level of the building, is enclosed with concrete block walls and a roof supported by a structural steel frame. The steel frame also supports the runway girders for the crane which services the area.

The roof of the building consists of a concrete slab roof and a 20-year built-up roof. An elevator provides access to the lower floors of the building, and a stairway is provided as an alternate access.

3.5 Biological Shield Cooling

It is necessary that cooling be provided in the concrete opposite the reactor core.

This system is designed so that maximum cooling effectiveness is achieved at 5-10 inches in from the inside face of the concrete wall. Thermal gradients in the concrete will not exceed 15°F per foot. The total gradient through the wall is less than 50°F. The shielding cooling system is designed for a duty of 2.5×10^6 BTU/HR.

3.6 Suppression Pool Cooling

A cooling coil is installed in the suppression pool. This will permit controlled cooling of the suppression pool after an incident.

3.7 Heating and Ventilation

The design of heating and ventilating systems is generally in accord with conventional practices. The nuclear system does, however, require some special considerations. The basic parameters and requirements are listed below:

Ventilation - Reactor Building

General Areas

The following criteria apply to the ventilation system in the reactor building exclusive of the dry well and suppression chamber:

1. Temperature limits:
Accessible areas - 100°F max.
Inaccessible areas - 150°F max.
All areas - 50°F min.
2. Air requirements:
inlet filtering not required.
Exhaust filtering of general building air not required.
3. Air flow:
Recirculation of air from potentially contaminated areas is not permissible.
Flow shall be generally toward areas which are most active or most liable to contamination.
All exhaust flow shall be directed to the stack.

3.7.1 Refueling Building Ventilation

The ventilation of the refueling area during normal plant operation shall be handled as a part of the conventional building ventilation. During refueling operations the exhaust from this area shall be directed through high efficiency filters to the stack. This air shall be maintained at a negative pressure with respect to atmosphere.

3.7.2 Dry Well Ventilation

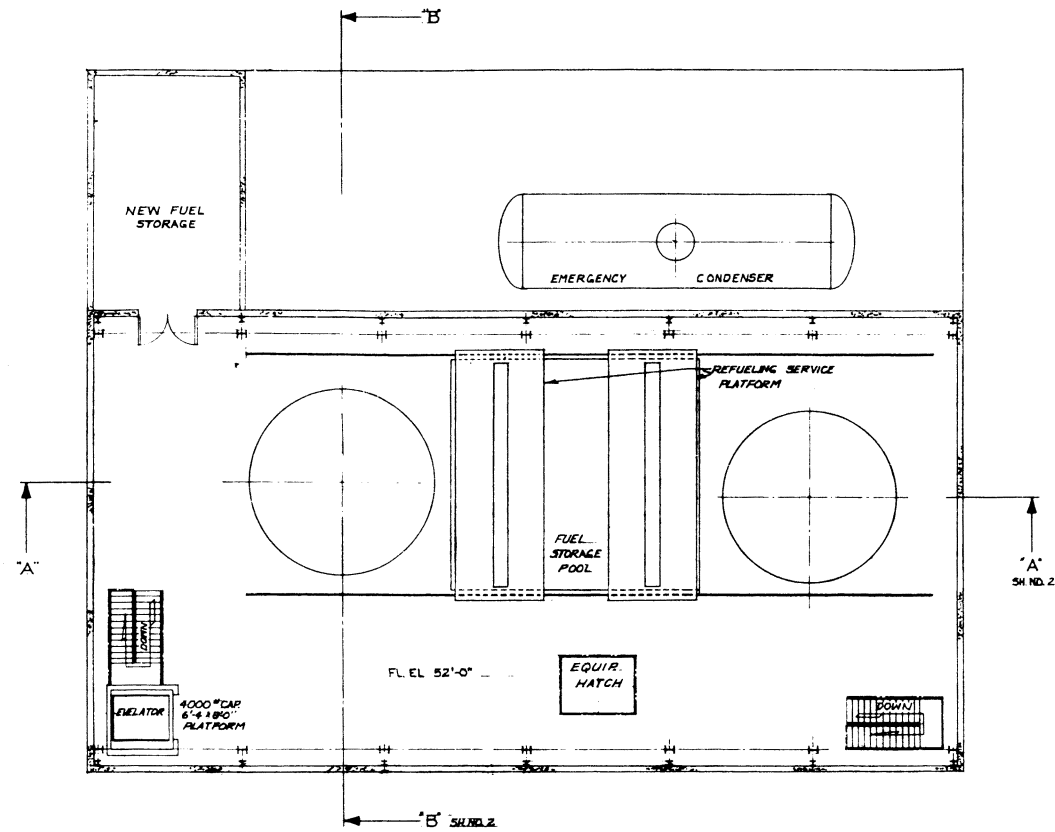
A closed forced recirculation system employing coolers is used to provide cooling in the drywell. This system is designed to control the air temperature in the drywell to a maximum of 150°F during normal plant operation.

A single unit cooler above the reactor vessel head will provide cooling in this area. The drywell is separated into two areas by the vessel to drywell wall seal but some cooling interchange will be provided through the large vent pipes.

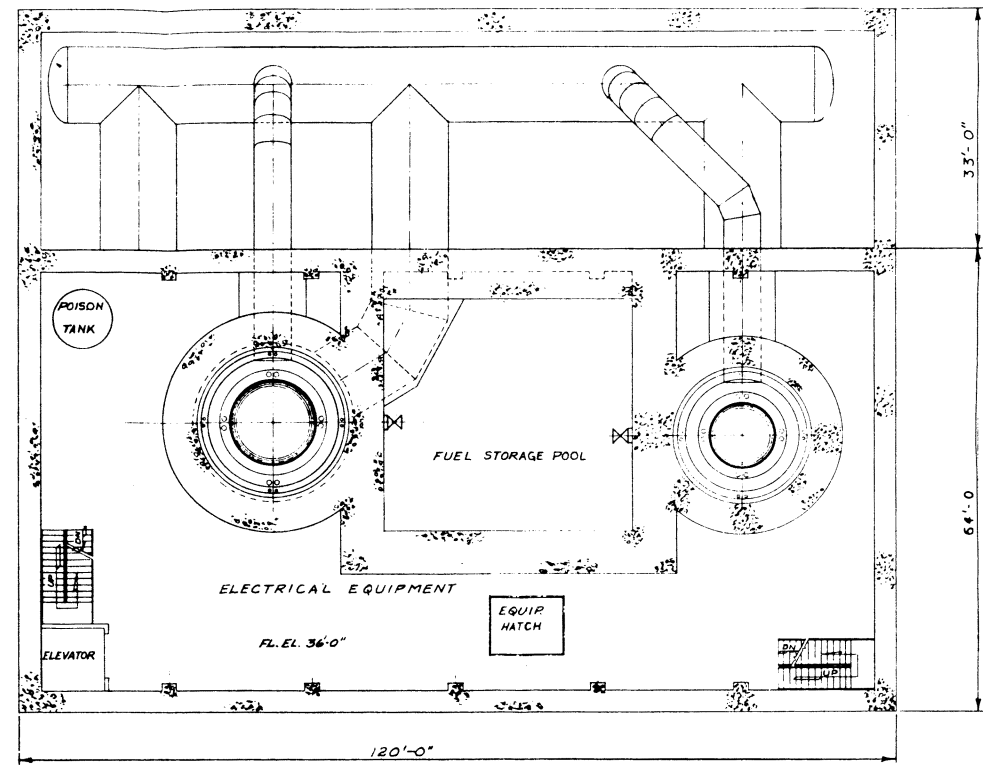
The ventilation system includes valved connections which will permit purging of the drywell to the stack at shut-down prior to access to the drywell.

3.7.3 Suppression Chamber Ventilation

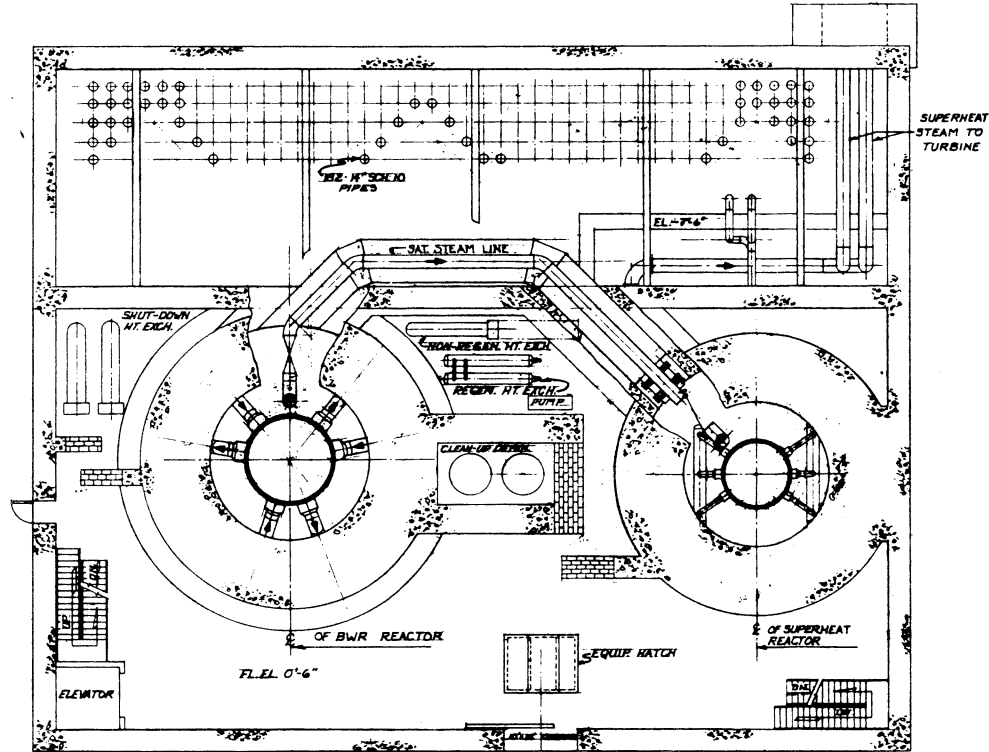
The suppression chamber above the suppression pool will not normally be ventilated. Provisions are incorporated to purge this area under controlled conditions to the stack. An in line filter with bypass provisions is provided in the exhaust purge line. The filter will be a high efficiency filter so arranged that disposal handling can be conveniently accommodated.



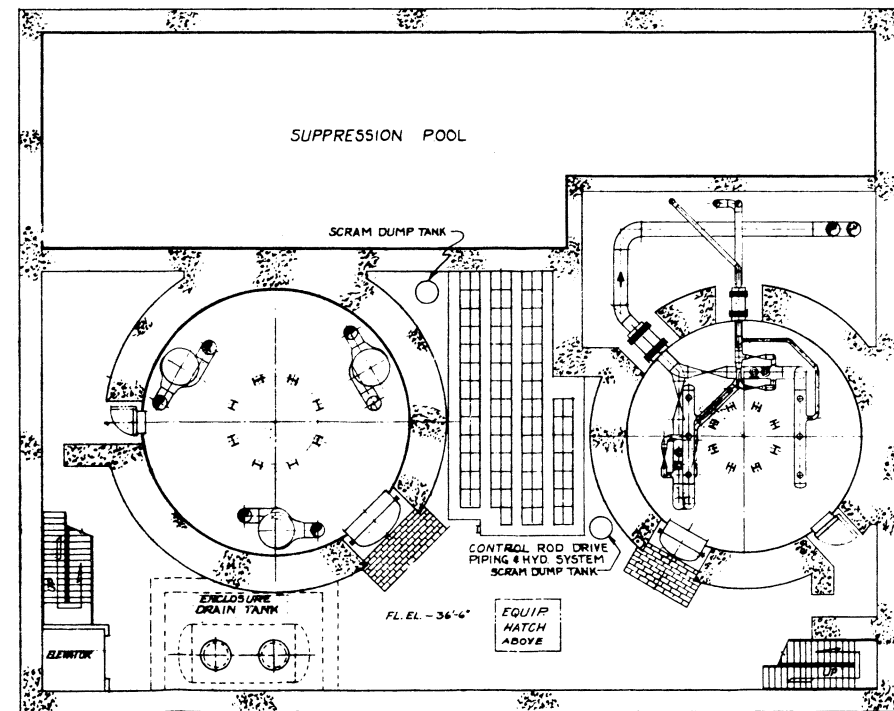
PLAN No. 1



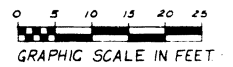
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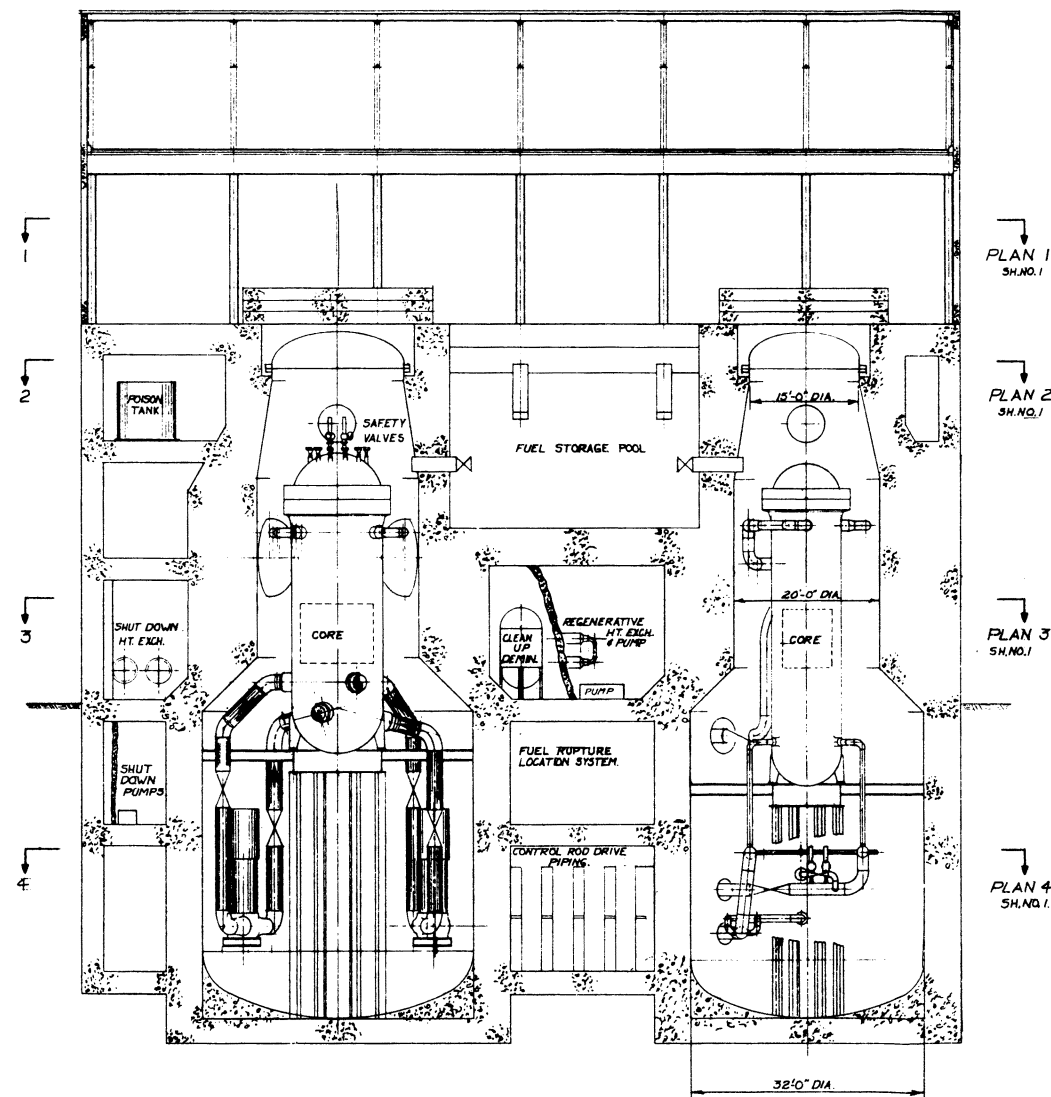
PLAN No. 3



PLAN No. 4



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			REACTOR ENCLOSURE ARRGT.
			SCALE FOR
			MCAP 3003
			198E124
			SH. NO. 1



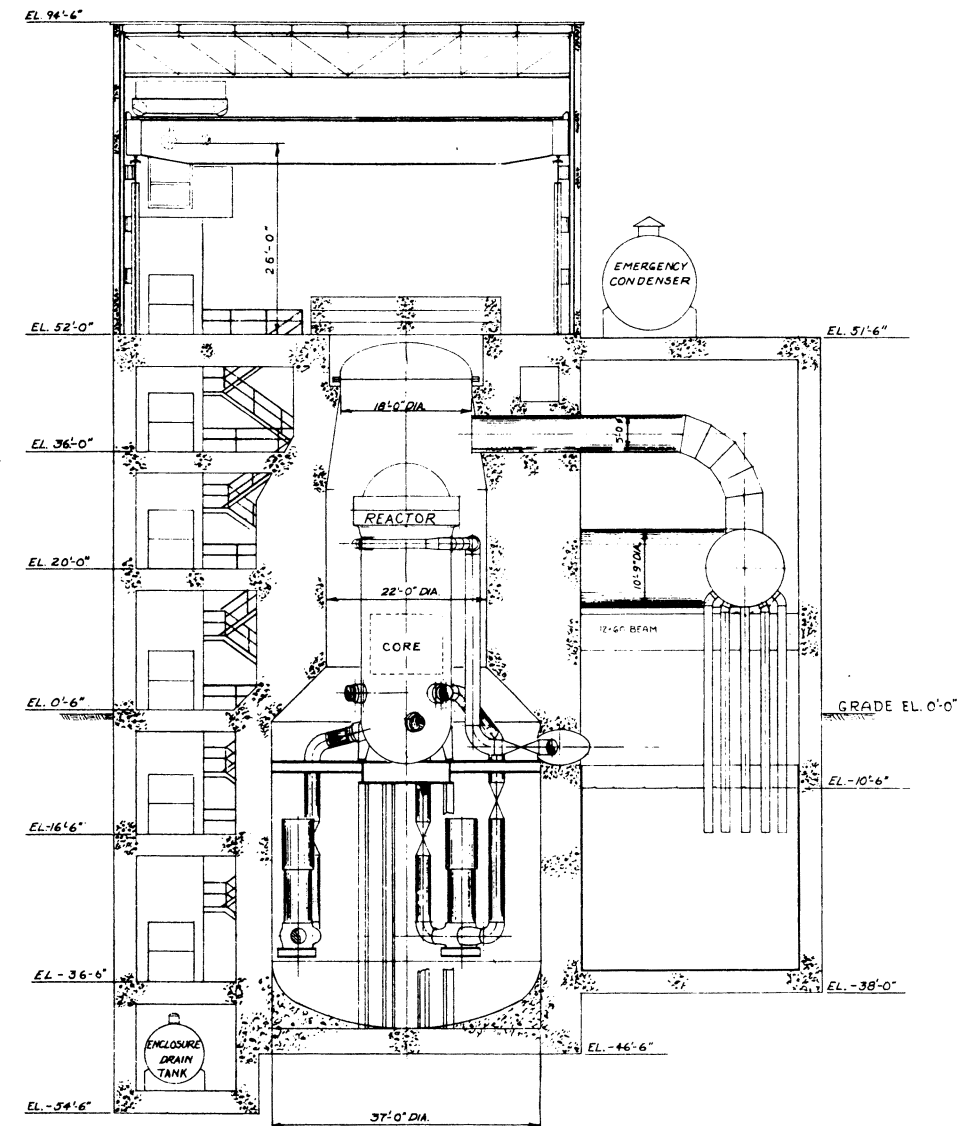
ELEVATION "A-A"

PLAN 1
SH. NO. 1

PLAN 2
SH. NO. 1

PLAN 3
SH. NO. 1

PLAN 4
SH. NO. 1



ELEVATION "B-B"



REVISIONS

DRAWN BY H. J. HARRIS 11-3-60	APPROVED		ATOMIC POWER EQUIP. DEPT. SAN JOSE, CALIF.
	BY	FOR DATE	
CHECKED BY R. J. HARRIS 11-3-60			TITLE STUDY REACTOR ENCLOSURE ARR'GT SH. NO. 1 PLANS, 3000 & 3001 MADE FOR MCAP 3003
DESIGNED BY H. J. HARRIS 11-3-60			SCALE 1" = 10'-0"
			198E124
			DWS. NO. SH. NO. 2
			REV.

4.0 Reactor Auxiliary Systems

The following systems are included as reactor auxiliary systems:

- (1) Reactor By-pass Demineralizer System
- (2) Shutdown Cooling System
- (3) Fuel Pool Cooling System
- (4) Emergency Cooling System
- (5) Control Rod Feedwater Supply System
- (6) Reactor Core Spray System
- (7) Liquid Poison System
- (8) Superheat Moderator Preheat System
- (9) Nuclear Steam Supply System Piping and Valves

4.1 Reactor Bypass Demineralizer System

The reactor coolant is maintained at a purity of one megohm of resistivity or better by utilization of the reactor bypass demineralizer system. Low solids are required for the following two reasons:

1. To maintain good heat transfer conditions in the core by eliminating the possibility of solids deposition in the fuel elements.
2. To minimize the concentration of activation products and fission products in the reactor coolant by continuous removal. Solids entering the reactor as products of corrosion-erosion become irradiated in the high flux of the core and thus become secondary sources of beta and gamma radiation.

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.

The reactor bypass demineralizer is a closed loop system. Reactor water is taken from the reactors and passes to the bypass system recirculating pump, the tube side of the regenerative heat-exchanger, the nonregenerative heat-exchanger, the mixed bed demineralizer, the shell side of the regenerative heat exchanger and back to the reactor feed piping.

No chemical regeneration of spent resins is 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 contaminated resin hold tank to be held for off-site disposal.

Fresh resins are hydraulically transferred to the cleanup demineralizer from the resin storage tank in the condensate demineralizer system. During initial start-up operations, used cleanup system resins may also be hydraulically transferred to this storage tank for dirt removal or regeneration.

A simplified flow diagram of the reactor by-pass demineralizer system is shown on the following page. A complete reactor by-pass demineralizer piping and instrument drawing is included at the end of this section.

The reactor by-pass demineralizer system is sized for a continuous by-pass of 140,000 #/hr of reactor water.

4.2 Shutdown Cooling System

The shutdown cooling equipment primarily provides for decay heat removal from the reactor water system prior to and during the refueling operation. In a normal shutdown for refueling, the reactor is initially cooled at a controlled rate by passing decay heat-produced steam through the steam bypass system to the main condenser. After approximately three hours of controlled bypass flow and system temperature reduction, the shutdown system is put into operation. At this time, the reactor decay heat output is near the rated duty of the shutdown system, and the reactor water has been cooled to 125°F and afterwards held at or below this level for the refueling operation.

The system consists of parallel cooling circuits, each with a pump, heat exchanger, related piping and instrumentation.

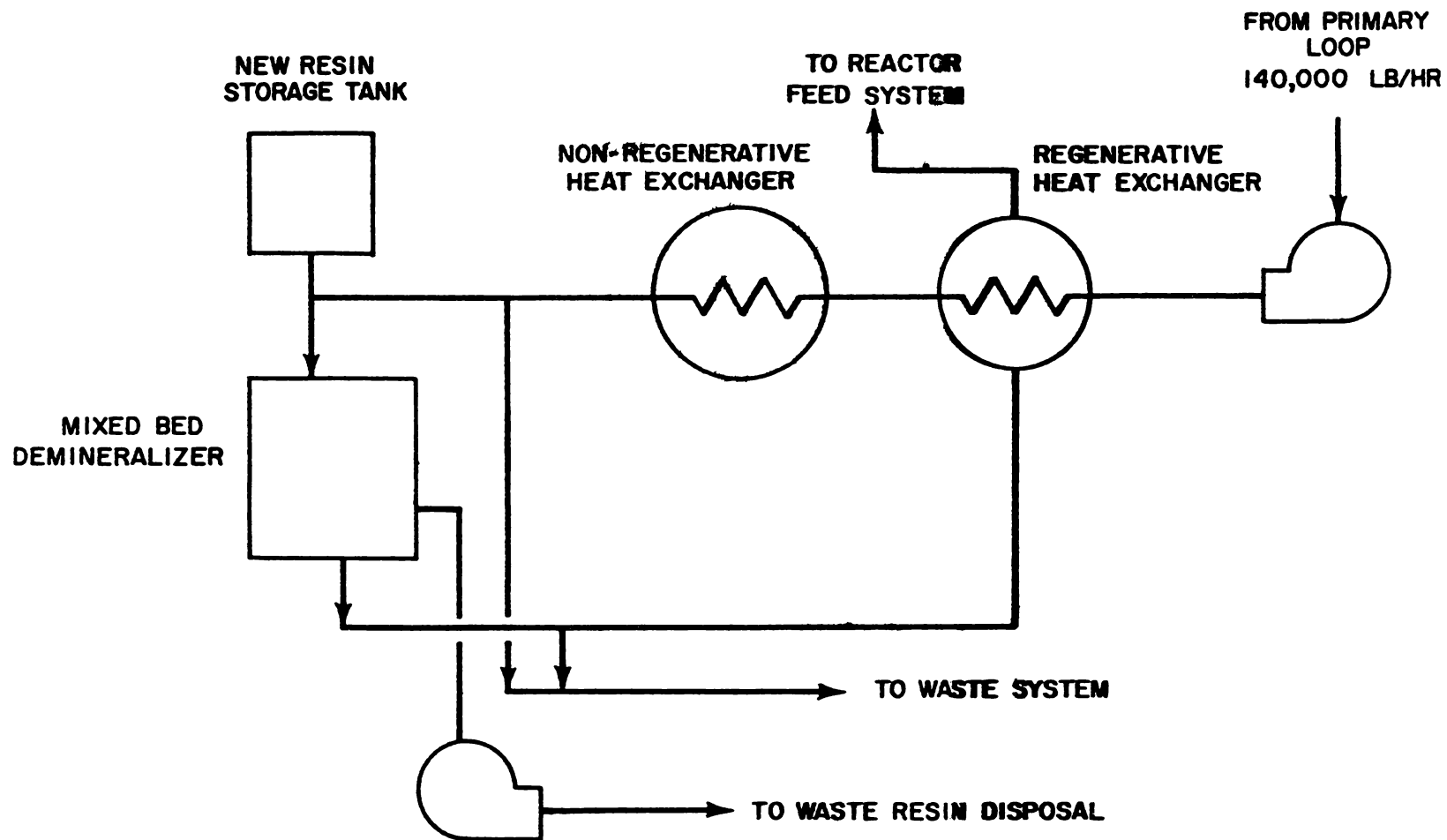
During normal plant operation, one of the system heat exchangers serves as a refueling pool cooling source in conjunction with the refueling pool circulating pumps.

If an emergency condition occurs which requires a reactor scram and use of the emergency condenser, the reactor may be shutdown in a similar procedure as for normal refueling preparation, except that the heat removal by the emergency condenser would result in a longer initial phase of operation. In this case, the shutdown system would be utilized after approximately eight hours of emergency condenser operation. At this time the shutdown system would be able to cool the reactor as in a normal shutdown.

The shutdown cooling system is sized for 1 percent of the combined thermal power of the boiling water reactor and the superheat reactor. The shutdown cooling system is shown on the nuclear steam supply piping and instrument drawing. This drawing is located at the back of this section.

4.3 Fuel Pool Cooling System

The fuel pool cooling system is required to remove the heat generated by spent fuel elements which have been removed from the reactor and are awaiting shipment to a reprocessing plant. The system includes circulating pumps, in-line filters, and a piping and valve system which interconnects the fuel pool, the reactors, and the shutdown heat exchanger system. This latter system serves as the fuel pool heat exchanger.



REACTOR CLEANUP SYSTEM

Pool water temperature is maintained at or below 125°F by circulating pool water through one of the shutdown heat exchangers. In addition, the system is designed to provide a circulating system for the pool and reactors during refueling operations. At this time, water is delivered to the pool and the sealed reactor dry well are above the reactor vessels. This water is circulated through the reactor and, via the shutdown heat exchanger system, back to the fuel pool circulating pump. Thus, the system provides additional circulation and temperature control of the water shielding required during refueling.

Two pool circulating pumps are provided, each is equipped with appropriate filters, valving and instrumentation.

4.4 Emergency Cooling System

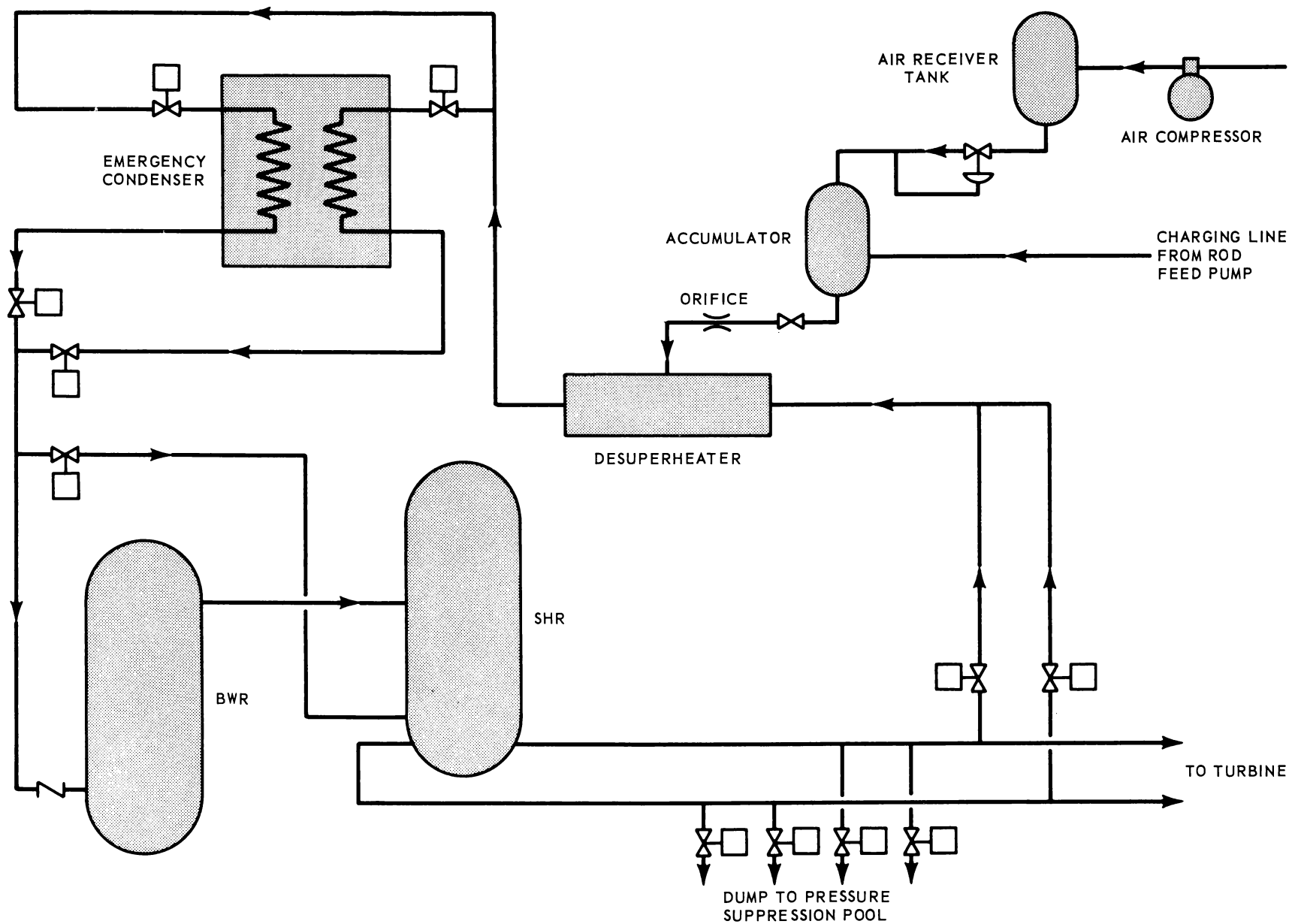
The function of the emergency cooling system is to transfer decay heat from the reactors to prevent overpressure in the primary system and to prevent excessive temperature rise in the fuel cladding in the event the main condenser should become isolated from the primary system. The conditions under which 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 bypass valves to open and pass steam to main condenser
3. Containment isolation valve closure
4. Main condenser failure

The emergency cooling system consists of high capacity dump valves which pass steam to the pressure suppression vent pool when the system is initially brought into service; a desuperheater and its associated equipment to remove superheat from the initial steam flowing to the emergency condenser, an emergency condenser vented to the atmosphere, and the necessary piping, valves, and controls. A simplified emergency cooling system flow diagram is located on the next page.

At the onset of the emergency condition, an automatic signal scrams both reactors and simultaneously activates the emergency cooling system. The dump valves, emergency condenser, and desuperheater coolant valves are immediately opened which results in initial steam flow rates of approximately 60% of full load steam flow. These high flow rates are required in order to assure that the superheater cladding will not exceed desirable limits. 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 and the emergency condenser is utilized to bring the system down to atmospheric pressure.

The dump valves, having flow capacity of approximately 60% 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 dry well, which pass steam to the pressure suppression vent pool. The valves are fast acting, closed to fully open within approximately one second,



EMERGENCY COOLING SYSTEM FOR 300 MW AEC STUDY

in order to prevent a substantial drop in steam flow through the superheater during the initial transient condition.

A desuperheater is provided in the steam lines to the emergency condenser to desuperheat the initial steam and thus prevent initial high thermal shock to the condenser tubes. The desuperheater system consists of a high pressure accumulator, fast acting valve in the coolant line from accumulator to desuperheater nozzles, and the desuperheater. When the emergency condenser steam line is initially opened, the desuperheater will be simultaneously brought into service. The desuperheater coolant lines and accumulator will be sized to remove all superheat from the initial steam flow to the condenser. 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 accumulator pressure drops to primary system pressure, the emergency condenser will be adequately warmed and capable of accepting superheated steam.

The emergency condenser is sized to be able to condense approximately six percent of rated steam flow. During normal plant operation, the shell side of the condenser is filled with water at atmospheric pressure. The tube bundles are connected to the steam lines through normally open inlet valves. The emergency condenser is brought into service by either an automatic signal or by manual operation of the discharge valves. The condenser is at the highest practical location to provide sufficient gravity head for natural circulation of the coolant.

4.5 Control Rod Feedwater Supply System

The control rod feedwater system supplies high pressure demineralized water to the control drive system to provide a seal leakage flow of water into the reactor for the drive mechanisms and to provide cooling for the drive unit.

The system is made up of two high-head low-flow pumps, filters, control valves and process instrumentation. Each pump is capable of supplying the requirements of the rod drive systems. Normal operation consists of taking pump suction from the feedwater system prior to the lowest pressure feed heater. At this point, the feedwater is below the maximum temperature permitted by the rod drive mechanism. An alternate suction line is provided to allow pumping from the condensate storage tank.

During normal operation, pump flow is delivered to the rod drive control system. An alternate discharge connection to the feed lines of the reactors is provided so that these pumps may be used as emergency feed pumps if necessary. The pumps will be connected to the emergency power source and the supply lines from the pumps are sized to take the flow of both pumps.

4.6 Reactor Core Spray Water System

To guard against possible fuel melting in the event of a loss of water accident, a core spray system has been provided. The system supplies cooling water to the reactor core spray and liquid poison sparger previously described. Two independent, full capacity pumping systems are provided. Upon receipt of an initiating signal from the control and instrumentation

system, one of the pumps is automatically started. If for any reason the on-line pump should fail, the second unit is automatically brought into service. The systems take suction from the plant condensate storage tank and continue to operate until temperatures in the reactor have been reduced to safe levels or until the water level in the reactor has been restored. Connections are provided in the piping system for utilizing water from the plant fire mains if necessary.

4.7 Liquid Poison System

In the event of an equipment malfunction which prevents the control rod system from shutting down either reactor, the operator may inject a sodium pentaborate solution into the system. This solution is injected through previously described spargers, mixed with the water in the vessel and, because of the high neutron absorption cross-section of boron, causes the reactor to go sub-critical. The system consists of a storage tank, a set of positive displacement pumps and appropriate piping, valves and instrumentation. Three pumps are included; two pumps have a-c electric motors, and the third pump has a d-c motor. This system is intended as a back-up control to be used only at the discretion of operating personnel. Initiation is, therefore, manual from the reactor control room. Once the operator decides to energize the system and pushes the button, about 30 seconds are required to inject sufficient pentaborate into the reactor and shut it down.

4.8 Moderator Preheat System for Superheat Reactor

Prior to superheat reactor start-up, it is desirable to raise the temperature of the moderator to near saturation. The moderator preheat system utilizes the boiling reactor clean-up loop. As a source of hot water, clean-up loop water is delivered to the superheat reactor moderator section through a line from the clean-up pump discharge. Moderator water is returned to the boiling reactor vessel by means of a transfer pump. The moderator preheat loop flow is controlled by the level control systems of the two reactors. The moderator transfer pump flows are matched with the clean-up circulating pump flow. The proposed system is designed to preheat the moderator volume in approximately two hours using full system capacity.

4.9 Nuclear Steam Supply System Piping and Valves

The piping and valve design and selection meet the requirements of Section I of the ASME Boiler and Pressure Vessel Code and also conform to Section I of the American Standard Association Code for pressure piping and related code cases.

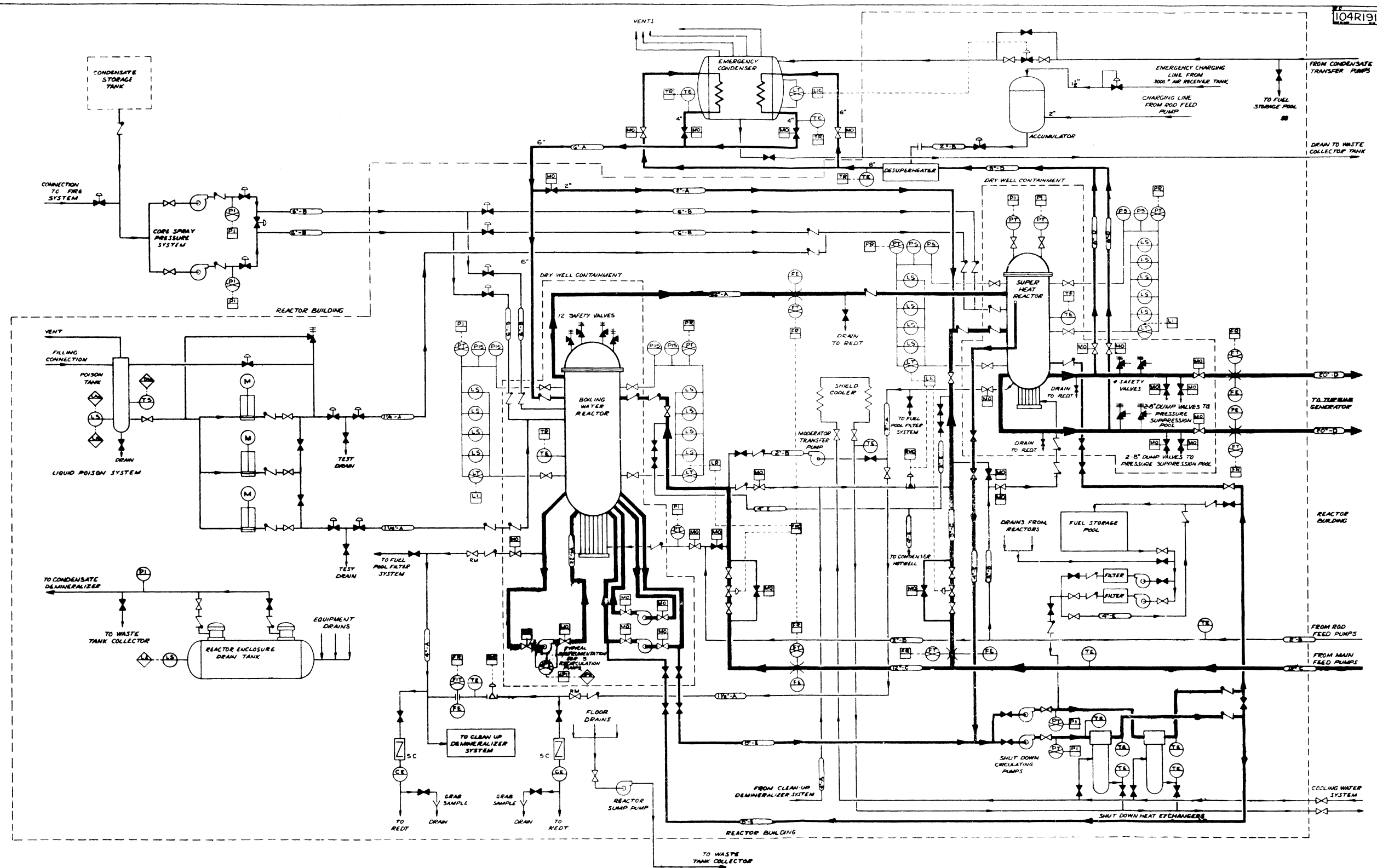
4.9.1 Piping

The Nuclear Steam Supply System P & ID indicates the piping in a diagram form including size and major valves requirements. The Pipe Material Tabulation Drawing provides the piping material selection and pipe schedule as referenced in the piping diagram.

4.9.2 Valves

Valve selection and design are based on the following requirements:

1. Valves indicated in P & ID for containment and/or system isolation are gate type with remote motor operators. Valves have butt welded ends. Body materials conform to the system piping material specification.
2. The bonnet is a bolted type with provisions for seal weld. The body-to-bonnet joint shall be leakage free when tested under hydrostatic pressure.
3. Stem seal stuffing box design shall utilize two sets of packing, each of which shall be able to withstand design conditions without visible leakage. A lantern ring with a bleed off connection shall be installed between each set of packing. The packing material shall be mica impregnated asbestos with monel wire insert or equivalent.
4. The electric motor operator is dri-proof rated at 110 V a.c., 3 phase, 60 cycle. Containment isolation valves are tied into the DC power supply through a control inverter unit.
5. Valve position indicator lights for fully open and fully closed positions are furnished.



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PIPE MATERIAL

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CONT ON SHEET 2 SH NO. 1

- Ref. A These lines shall be fabricated from alloy steel ASTM A-376 type 304.
The minimum walls shall be as follows:

<u>Pipe Size</u>	<u>Min. Wall Thickness</u>
1" thru 18"	Schedule 80
20" thru 24"	Schedule 100

The joints shall be butt welded for sizes $1\frac{1}{2}$ " and over, and
socket welded for $1\frac{1}{4}$ " and under.

- Ref. B These lines shall be fabricated from Carbon Steel, ASTM A-106 Gr. B.
The minimum walls shall be as follows:

<u>Pipe Size</u>	<u>Min. Wall Thickness</u>
2" thru 12"	Schedule 80

The joints shall be butt welded for sizes $2\frac{1}{2}$ " and over, and
socket welded for 2" and under.

- Ref. C. These lines shall be fabricated from Carbon Steel ASTM A-155 type
KC 70. The minimum walls shall be as follows:

<u>Pipe Size</u>	<u>Min. Wall Thickness</u>
2" thru 16"	Schedule 80
26"	1.25"

The joints shall be butt welded for sizes $2\frac{1}{2}$ " and over, and
socket welded for 2" and under.

- Ref. D. These lines shall be fabricated from Alloy Steel ASTM A-155
Gr. $2\frac{1}{4}$ Cr. Class 1. The minimum walls shall be as follows:

<u>Pipe Size</u>	<u>Min. Wall Thickness</u>
2" thru 6"	Schedule 80
8" thru 10"	Schedule 100
26"	1.50"

All joints to be butt welded.

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Proposal
PIPE MATERIAL

CONT ON SHEET

SH NO. 2

Ref. E These lines shall be fabricated from welded carbon steel ASTM A-53 or ASTM A-120. The minimum wall thickness shall be as follows:

<u>Pipe Size</u>	<u>Min. Wall Thickness</u>
1" thru 12"	Schedule 40

The joints shall be butt welded or flanged for sizes $2\frac{1}{2}$ and over, and socket welded for sizes 2" and under.

5.0 Plant Control and Instrumentation

5.1 General Plant Control

The control and instrumentation system provides the necessary information and control to enable 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 and associated 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. These components have been thoroughly proven on numerous reactors installed throughout the world.

5.2 Plant Operation Control

5.2.1 Control System Requirements

The control system for this two reactor power plant will be designed under the following requirements:

1. Reactor power of either the boiling water reactor or the superheat reactor will be controlled by positioning the control rods.
2. System pressure will be controlled automatically by the initial pressure regulator on the turbine control valve.
3. System steam temperature will be controlled by adjustment of the superheat reactor power.
4. Partial or full load generator rejections will be absorbed by a full capacity bypass valve to the condenser in order to prevent the reactors from scrambling on high flux or high pressure.

5.2.2 System Pressure Control

An initial pressure regulator (IPR), in controlling the turbine control valve, maintains system pressure. With the turbine under this type of control system load demands must be observed by the reactor operator(s). On a load demand, the operator withdraws the control rods, reactor power increases, pressure increases, and the pressure regulated turbine control valve, in readjusting pressure, accepts the power generated by the reactor.

5.2.3 Reactor Control Under Load Changes

Normal load changes are performed in a stepwise manner between the boiler and the superheater in order to maintain close control of the superheater outlet temperature. On a load increase, the power from the boiler is increased a finite increment. Then the superheater power is increased to maintain the outlet temperature. On a load decrease, the procedure is reversed in that the superheater power is decreased enough to decrease the outlet temperature, and then a boiler power decrease raises the temperature.

Although these load changes are described as a series of steps in operating the boiler and superheater, it is expected that the operators, with training and experience, will be able to maneuver the two reactors

5.2.4 Bypass Steam

At the reactors, load rejections cause pressure to rise, and because of the positive effect of pressure, flux rises. A flux rise to the high flux level scrams the reactors. The bypass valve to the condenser serves to prevent such a scram by providing a path to the condenser for turbine rejected steam. With full capacity bypass valves that open as fast as the turbine control valves close, the reactor sees little or no load change

In addition to a fast response, the bypass valve must also regulate reactor pressure when it is in operation. During operation, it behaves essentially the same as the pressure regulated turbine control valve. This requirement insures that the valve bypasses the correct amount of rejected steam without causing the pressure disturbance

An additional requirement, that of accommodating a stop valve closure by the bypass valve without scrambling the reactor, is deemed necessary because in all probability a stop valve closure implies a major turbine incident (loss of oil pressure, overspeed, loss of vacuum) that would require a plant shutdown. Thus, on a stop valve closure, the bypass valve is required to operate fast enough to prevent a system overpressure and to provide a path for boiler steam flow through the superheater to the condenser. For example, the stop valve closes, pressure rises, the reactors scram on stop valve closure, high flux, or high pressure, and the bypass valve opens on a stop valve closure signal to shunt excess steam to the condenser and maintain system pressure.

5.2.5 Start-up and Shutdown Procedure

The start-up and shutdown procedure for the separate superheat reactor plant is outlined below:

Start-up After Refueling

1. After a refueling shutdown both the boiler and superheater will be cold and flooded.
2. The superheater elements will be drained.
3. The water level in each reactor will be lowered to the normal water level.
4. With the superheat reactor still subcritical, the boiler is brought to criticality and the moderator heating process begins.

5. The superheater is heated by circulating the boiler moderator through the superheater.
6. The boiler continues to operate at low power, heating the moderator and boiling to increase steam pressure.
7. After all sections of the superheater have been cleared, the boiler is brought to a sufficient power level to provide 10%-20% steam flow through the superheater bypass valve to the main condenser.
8. With 10%-20% steam flow established, the superheater can then be made critical.
9. The boiler and superheater can then be brought up to operating pressure and temperature.

Normal Start-Up

After most shutdowns, the superheater fuel elements will already be cleared of water. Then, as soon as boiler steam flow is established the superheat reactor can be made critical and brought to operating steam temperature.

Normal Shutdown

The normal shutdown procedure is as follows:

1. The power level in both the boiler and the superheater is reduced manually.
2. The bypass valve to the main condenser is opened and the turbine removed from service.
3. If the shutdown is only for a period of hours, the reactors are continually cooled by passing steam to the main condenser.
4. If the shutdown is for a longer period of time, the reactors are made subcritical and are cooled for several hours by passing steam to the main condenser.
5. The superheater cools itself by radiating heat directly from the fuel elements to moderator. The moderator of all reactors is cooled by the shutdown heat exchanger.
6. If the vessel head is to be removed, the moderator water level is raised to flood the superheater fuel elements. If the vessel head is not removed, the superheater fuel elements will remain unflooded.

5.3 Reactor Protection System

The protection system for the boiling water reactor and the superheat reactor 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 scram of either or both reactors originate from a variety of control and detection devices. These signals activate control circuitry to cause the reactor to be immediately shutdown and initiate operation of penetration closures, emergency cooling, and various other devices which are necessary for safe plant shutdown.

Each reactor has an independent two channel, "fail-safe", reactor protection system which must be de-energized to produce a reactor shutdown or other safety system function. The protection systems for each reactor are appropriately interconnected to provide a complete system. The total system 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.

Whenever practical, the two channels of each subsystem are physically separated and clearly identified so as to minimize the possibility of maintenance personnel causing an accidental shutdown.

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 it should. Furthermore, any logic element or power switch may be removed from the system without impairing the integrity of the reactor protection system.

Preliminary analysis of the reactor system indicates that the following functions or type of functions should scram the reactors:

Scram from Short Reactor Period (BWR or SHR)

This prevents the reactor from increasing power too rapidly, thus anticipating a too-high flux condition. The reactors will scram independently for the following reasons: (1) Short period protection will be used on the BWR only up to the power point where boiling begins. Period trip is then bypassed. During this time, the SHR may either be shutdown or just critical, and there should be no thermal problem should the BWR scram when there is a small amount of steam flow. (2) The SHR is brought up in power only when there is cooling steam flow from the BWR. As in the BWR, there will be period protection only up to the point where boiling begins in the SHR moderator. This power point should be low enough so that the amount of superheating is low and a scram of the SHR will not cause too-rapid cooling of any part of the system.

Scram from High Neutron Flux (BWR or SHR)

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

Scram from High Reactor Pressure (BWR or SHR)

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 for cooling (open valves to pressure suppression pool and emergency condenser) are initiated.

Scrams from High Dry Well Pressure (BWR or SHR)

The rise in pressure indicates a rupture within the dry well severe enough to drain the water in the reactor vessel to the point of uncovering the core if no corrective action is taken. This would cause the release of radioactive material in the dry well. Scramming the reactor minimizes the possibility of damaging the core. Since there is the possibility of release of radioactive material, isolation of the dry well is initiated by closing the steam isolation valves, and since steam flow is interrupted, emergency cooling is also initiated.

Scram from Low Water Level in Reactor Vessel (BWR or SHR)

The same reasons and actions as for High Dry Well Pressure apply.

Scram from High Water Level in Reactor Vessel (SHR)

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

Scram from High Water Level in Drive System Scram Dump Tank

This insures that the reactors are not operated with insufficient free volume in the drive system scram dump tank which would prevent rapid insertion of the rods (scram).

Scram from Low Condenser Vacuum

This insures that the reactors are not operated without their main heat sink. Continued operation would cause escape of radioactive steam to the turbine room through ruptured relief diaphragms.

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.

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 bus. Large inertias built into the M-G sets allow a few seconds auxiliary power interruption without scram. The fail-safe feature initiates steam isolation valves closure, which then initiates cooling.

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 reactors. Emergency cooling is also initiated since steam flow is interrupted.

Scram from High Steam Temperature in SHR

Outlet steam temperatures will be measured in the steam channels. An excessively high temperature will cause scram since it indicated insufficient cooling in the channel which can cause fuel element failure.

Scram from High Ratio of Flux to Steam Flow in SHR

This anticipates a dangerous rise in temperature, and therefore, would precede a High Temperature Scram.

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.

Typical block diagrams applicable to the boiling water reactor and to the superheat reactor are shown on Drawing 612D731. Each channel of the system is powered by two 24/12 volt D-C power supplies, either one of which can provide power for the load. The logic elements are composed of diodes arranged to perform a desired logic and to provide a dual series element transistor 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 inrush current. During normal operation the input to the power switches is held at a nominal 12 volts by the logic elements. To initiate reactor shutdown, the input to the power switches is dropped from 12 volts to a fraction of a volt, causing all controlled rectifiers to cut off. Actually, the trip level of the power switches is established at about 8 volts. The typical input to each logic element is a nominal 12 volts D-C supplied through a pressure or level switch or supplied directly from one of the neutron monitoring instruments. If all inputs are 12 volts, all logical element transistors are held in their conducting state and the input to the power switches is maintained at 12 volts D-C. If any single output 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 scrammed.

SAFETY SYSTEM FUNCTIONS

	<u>Scram</u>		<u>To Emergency</u>	<u>Open Dump</u> <u>Valves to</u> <u>Pressure Sup-</u> <u>pression Pool</u>	<u>Close to</u> <u>Steam</u> <u>Isolation</u> <u>Valves</u>
	<u>BWR</u>	<u>SHR</u>	<u>Condenser</u>		
Short reactor period, BWR	X				
Short reactor period, SHR		X			
High reactor flux, BWR	X	X			
High reactor flux, SHR	X	X			
High reactor pressure, BWR			X	X	
High reactor pressure, SHR	X	X			
High dry well pressure, BWR	X	X	X	X	X
High dry well pressure, SHR	X	X	X	X	X
Low water level, BWR	X	X	X	X	X
Low water level, SHR	X	X	X	X	X
High water level in SHR	X	X			
High water level in scram dump tank	X	X			
Low condenser vacuum	X	X			
Closure of both turbine stop and bypass valve	X	X	X	X	
Loss of auxiliary a-c power	X	X	X	X	X
Closure of steam isolation valves	X	X	X	X	
High temperature, SHR	X	X			
High ratio-flux to steam flow, SHR	X	X			

5.4 Reactor Neutron Monitor System

An instrumentation system is provided to monitor the neutron level of the reactor from startup through full power. The instrumentation will cover a range of 9 decades in three phases:

1. Startup - 3 channels
2. Period or Intermediate Range - 3 channels
3. Flux Level or Power Range - 6 channels

5.4.1 Startup Range

With the initial fuel loading, a neutron source is inserted in the core to assure a count rate of several neutrons per second.

The startup instrumentation covers the range upward to about 10^7 counts per minute. Three channels of instrumentation monitor this phase of the operation. The primary neutron detectors are proportional counters housed in guide tubes which place them adjacent and external to the reactor vessel wall about two feet above the midplane 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 series of pulses proportional to the reactor neutron count level. 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 measurement. The count rate is recorded continuously. An additional circuit differentiates the log count rate and indicates the reactor period at this low level on a period meter. A short period at this low level is annunciated.

5.4.2 Period Range

This phase of the instrumentation system is concerned with the rate at which the neutron flux is increasing and 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 housed in guide tubes near the reactor similar to the startup channels. As the neutron count increases to about 10^7 cps the time average of this intensity of radiation results in a d-c current proportional to the neutron flux. This measurement is known as Log N and d-c techniques are required to measure these currents of about 10^{-10} amps. The Log N channels will be on a scale and overlap the previous startup channels. 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. A warning will be indicated visually and audibly if either channel approaches a short reactor period. Control rods are inserted if the reactor exceeds the short period trip setting.

5.4.3 Flux Level or Power Range

As the reactor reaches about 10^{-1} 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 will be functioning and will monitor the reactor through full power. These are six channels of instrumentation to cover this range.

The primary detectors are gamma compensated 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 flux amplifier of fast response and stable characteristics. The power level is read on a meter indicating the per cent of full power and is continuously recorded. A warning will be annunciated if any of the six flux level indicators exceed a preset percentage of over flux. A scram signal to insert rods and shut the reactor down is indicated whenever two of the six power level indicators show an excess of neutron flux (one odd and one even numbered channel is needed for a scram).

A similar system to monitor the superheat reactor is provided with two startup, two Log N/period, and six power range channels to monitor flux from startup to rated power.

5.5.1 In-Core Flux Monitor System

An extensive in-core flux monitoring system is provided to measure the flux inside the reactor core and provide operational data. The incore system does "see" more into the core center than does the external viewing system described above. This additional information concerning the flux in the core center helps to control the power density distribution and provides more accurate fuel power history in order that optimum scheduling and utilization of the fuel can be achieved.

The system will consist of chambers located between control rods and fuel bundles. In each position there will be four detectors at different elevations. The detector assemblies will be distributed throughout the core. A special cable will be provided to supply power and signal for the detector.

Only stainless steel is exposed to the reactor coolant. A calibrating wire utilizing the irradiation/counting technique is also part of the assembly.

The assemblies described above are enclosed in a thimble and penetrate the bottom of the reactor vessel. A special seal will be provided to fit the reactor vessel.

The detector ionization current provides an electrical signal which is read out through an amplifier and meter displayed in the main control room in a display integrated with the control rod position so as to be meaningful to an operator.

The instrumentation needed to count wires for the in-core monitor calibration is also provided.

5.5.2 In-Core Steam Temperature Monitor System

The steam temperature of the superheat reactor fuel assemblies will be measured and read out to provide the basic control and monitoring system. The steam temperature signal will be amplified, recorded, and displayed in the control room.

5.6 Control Rod System Instrumentation and Control

5.6.1 Rod Selection and Control

The boiling water reactor has 69 hydraulically driven control rods and the superheat reactor has 24 hydraulically driven control rods. Each control rod is controlled manually from the control room.

Selection of the one rod under 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 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 within the electrical circuit is responsive to manual position control.

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

Manual position control is accomplished with the use of one master position control switch which either energizes the "insert solenoid valve" in the hydraulic system to that particular rod selected for manual control.

5.6.2 Manual and Automatic Shutdown

Automatic rapid insertion of the control rods is accomplished in the dual fail-safe features of the Reactor Protection System by the off-normal condition of the various primary elements contained within the circuit.

A manual control for rapid insertion of control rods is also provided in the control room to be used at the discretion of the reactor operator.

Rod scram, when initiated either automatically by the Reactor Protection system, or by the manual scram switch, is accomplished by de-energizing the solenoid pilot valves (supply port closed when de-energized) which supply air loading pressure to the spring-loaded air-powered (air to close) inlet and outlet shutdown valves.

When the solenoid pilot valves are de-energized, the air pressure holding the shutdown valves closed is removed from the valves, and they are immediately forced open by the stored energy in the springs and the difference in pressure existing across these valves. The outlet shutdown valves dumps the pressure on top of the rod piston to a dump tank, and the inlet scram valve supplies hydraulic pressure from an accumulator to the bottom of the rod piston which forces the control rod into the core. The system is grouped so that one pair of solenoid pilot valves control one pair of shutdown valves (inlet and outlet) which accommodate three control rods.

5.6.3 Hydraulic System Instrumentation

The operating condition of the control rod hydraulic system is continually 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.

5.6.4 Rod Position Indication System

The position indication system provides simultaneous digital indication of the position of each of the 76 control rods.

Position indicators, and "all-in" and "all-out" pilot lights are provided for each rod. These devices are grouped together and arranged on the control room panel in a pattern simulating the relative locations rods with respect to each other in the reactor core, and is integrated on the panel with the in-core monitor system.

5.7 Liquid Poison System Instrumentation

5.7.1 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; also two pump controls are provided by 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 of the indicator at the pumps located in the turbine building.

5.7.2 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 discharging poison to the reactor.

5.7.3 Process Instrumentation

The liquid poison system is continually monitored by process instrumentation to indicate operating conditions, standby conditions, and accidental leakage of system valves. The storage tank is equipped with a level indicator and a low level alarm.

5.8 Main Power Loop Instrumentation

5.8.1 Temperature

Metal temperatures of the pressure vessel outside wall will be sensed in places of interest to the operating and design groups. This information will be used to determine the maximum allowable rate of heating and cooling.

All temperatures will be recorded on the process control panel.

5.8.2 Liquid Level

Liquid level in the pressure vessel will be measured continuously by means of externally mounted differential pressure type sensing devices. The reactor water level will be controlled, recorded, indicated, and high and low level annunciated.

Liquid level switches on the pressure vessel will monitor low level. In case of extremely low water level the switches will energize the safety circuit.

5.8.3 Pressure

Pressure will be measured in the pressure vessel and transmitted electrically to the control room. At the panel, the transmitted signal will be recorded and indicated.

Pressure switches on the pressure vessel will be used to monitor for high pressure. On high pressure the switch will energize the safety circuits. Two switches are provided for each safety circuit.

Four pressure switches will also be mounted on the containment sphere and used to actuate the safety circuit on an unusual increase in containment pressure.

Differential pressure will be measured across each of the four recirculating pumps. This signal will be transmitted to an indicator in the control room.

5.8.4 Reactor Feedwater Instrumentation

Water level in the reactor vessel will be controlled by a three element level control system. This system uses the measurement of steam flow, feedwater flow and water level. Normally, the steam flow signal equals that of water flow.

Water level, feedwater and steam flow rates will be recorded in the control room.

5.9 Plant Auxiliaries Instrumentation

5.9.1 Emergency Cooling System Instrumentation

When the closing of isolation valves calls for action of the emergency condenser, a sequence timer will go into operation. First, the superheater dump valves will be opened and they will dump for a controlled interval. As the end of the interval approaches, the emergency condenser coil drain valves will be opened and steam will begin to flow through the condenser. After this action is started, the dump valves will be closed.

Level in the Emergency Condenser will be measured and controlled. A differential pressure transmitter will sense condenser level and transmit to a level indicating controller which will regulate the control valve in the feed line from the condensate storage tank.

Temperature of the coil drain lines will be measured and recorded. A temperature transmitter with sensor in the superheated steam line will open the desuperheater control valve and allow water to be fed to the desuperheater from the accumulator.

Accumulator pressure will be transmitted to the control room where it will be indicated. Should the accumulator pressure drop below a preset point, the rod seal will be automatically started. When the pressure drops below a lower preset point, a self-operated regulating valve will open to allow a high pressure air line to feed the accumulator.

5.9.2 Reactor By-Pass Clean-up System Instrumentation

Clean-up flow will be measured and recorded. A remote manual control station will be used to position the regulating valves. Sample coolers and conductivity cells will be furnished for both the superheater and boiler clean-up lines.

5.9.3 Emergency Poison System Instrumentation

Level in the liquid poison tank will be annunciated when either a high or low level is sensed by a level switch. Low temperature will also be alarmed. Poison pump output pressure will be sensed by pressure transmitter and indicated in the control room.

5.9.4 Shield Cooler Instrumentation

A thermocouple in the shield cooler discharge line will be used when setting the cooling flow through the shield cooler.

5.10 Fuel Element Rupture System

A fuel element failure is detected by monitoring the off-gas from the main condenser 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.

Two channels of instrumentation monitor the off-gas and serve two functions: (a) to detect a fuel element failure, and (b) to automatically close the off-gas line to the stack in the event of excess activity in the off-gas.

5.10.1 Fuel Element Rupture Detection (BWR)

A fuel element failure is detected by monitoring the off-gas from the main condenser 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. The gamma spectrum analysis is used to determine the extent of the failure.

Two channels of instrumentation monitor the off-gas and serve two functions: (a) to detect a fuel element failure, and (b) to automatically close the off-gas line to the stack in the event of excess activity in the off-gas.

5.10.2 Fuel Element Rupture Detection (Superheat)

A fuel element failure in the superheat reactor is detected by: (a) the fission iodine activity in the off-gas, or (b) by process of elimination by comparing the off-gas activity with the activity of the non-condensibles in the saturated steam from the boiling water reactor.

5.10.3 Fuel Rupture Location System (BWR)

If a break should occur in the cladding of a fuel element, the plant may continue to operate as long as the quantities of fission products released to the atmosphere do not exceed the maximum permissible concentration. If the number of failures increases so that these maximums are exceeded, then it will be necessary to shut down the plant and replace the failed assemblies. To enable replacement with minimum downtime, it is desirable to know during operation which elements have failed. The Fuel Rupture Location System is provided for this purpose.

The mechanical components of the system are a number of collector tubes located in the reactor core. Each tube samples the steam-water mixture at a given core location. These tubes are piped out through the reactor vessel and shielding to a valve bank in the sampling room.

Steam samples, each representative of the steam generated by a particular fuel assembly, are withdrawn from the pressure vessel and stripped of non-condensable gases in de-gassifiers. The gas is monitored by gamma spectrometry instrumentation to determine the fission gas activity. In this manner, fuel element failures can be located with respect to the assembly while the reactor is operating. Two channels of gamma spectrometry instrumentation are provided. Each channel monitors one-half of the reactor core.

5.10.4 Fuel Element Rupture Location (Superheater)

A sample of superheated steam is taken from the outlet of each fuel bundle and monitored, one sample at a time, with gamma spectrometry instrumentation for traces of fission products. An alternate to the gamma instrumentation under consideration is neutron-sensitive instrumentation that will detect and indicate the delayed neutron activity of the individual samples.

Both systems can be operated at or near full reactor power or at reduced power levels. The system with the highest sensitivity to small fuel ruptures will be used.

5.11 Plant 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 the norm. Certain non-radioactive processes are monitored to provide alarm in the event of contamination which requires action to prevent a significant amount of radioactivity from reaching the environs.

5.11.1 Air Ejector Off-Gas Monitor

Non-condensable gases from the main condenser are monitored to determine the gross activity being discharged to the stack. If the gross radioactivity becomes too high (according to pre-established standards), the instrumentation trips the off-gas valve closed to prevent environs contamination. Trips are provided for alarm at lower activity levels.

Two channels of gamma spectrometry instrumentation are provided. One served as a backup to the other in the event of temporary equipment failure. This instrumentation also serves for fuel rupture detection.

5.11.2 Air Ejector Off-Gas Monitor Additions for Operation of the Superheat Reactor

Samples of gas are withdrawn from the off-gas line immediately downstream from the air ejector and immediately downstream from the charcoal filter. One sample is monitored continuously to establish gross activity released to the stack. During periods of high activity the two samples are compared for iodine and noble gas activity levels to establish filter efficiency in removal of the various isotopes.

5.11.3 Stack Gas Monitor

An isokinetic probe is employed to collect a representative continuous flow sample of the gas discharged from the stack. 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. A filter is provided to collect particulate matter from the sample stream. The filter is checked periodically for contamination.

5.11.4 Liquid Process Monitors

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

- a. Reactor Enclosure Cooling Water
- b. Service Water Discharge
- c. Circulating Water Discharge
- d. Waste demineralizer Effluent
- e. Condensate Demineralizer Influent
- f. Condensate Demineralizer Effluent
- g. Reactor Cleanup Demineralizer Influent
- h. Reactor Cleanup Demineralizer Effluent

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

5.11.5 Emergency Condenser Vent

The gross gamma activity in the vent stack of the emergency condenser is continuously monitored to determine breakthrough of the radioactive steam to the atmosphere in the event of tube failure during operation. This unit relies on the activity of the Nitrogen-16 isotope being present at the detector in the event of breakthrough.

5.11.6 Saturated Steam Monitor

A sample of the saturated steam from the boiling water reactor is taken and the non-condensable gases are removed and monitored for the fission gas activity. The fission gas activity in this process is compared to the off-gas activity to determine the location (which reactor) of a fuel element failure and the operating limits (if required) that may be governed by large failures. This system needs to be operated only when fuel element failures are suspected.

5.11.7 Central Installed Areas Monitoring System

Gamma sensitive detectors shall be located throughout the plant, each arranged to monitor a specific area. The gamma flux level at each detector shall be indicated continuously in or near the station control room. Adjustable trips will provide annunciation in the event of a high gamma flux level at any particular detector. A multipoint recorder will permanently record the gamma flux levels on a continuous cycle basis.

Each detector-indicator unit will be designed to monitor the gamma flux level over three decades between 0.1 mr/hr and 10 r/hr with the particular range chosen to suit the predicted conditions for that detector. The detectors will operate within stringent specifications regarding gamma energy dependence, accuracy and drift.

5.12 Control Room Arrangements

The over-all plant control and safety systems are designed to make the plant safe, reliable, and easy to operate. A centralized system of plant control is provided. Local instrumentation and control boards are included, where necessary, to supplement the facilities in the central control room. Conventional power plant instrumentation and control are used as far as possible.

5.13 Portable Instrumentation

Radiation Survey Instrumentation

Portable radiation survey instruments are required to monitor alpha, beta, gamma, and neutron radiations throughout the plant. These radiation measurements are used primarily to establish time and exposure limits for personnel entering radiation zones. The instruments are also used in connection with equipment de-contamination and non-routine process monitoring.

Particle Collection

Semi-portable air flow samplers are required to collect particulate matter on stationary filters. These are used in and about the plant for routine and non-routine air sampling.

Personnel Monitors

Personnel monitoring instruments such as beta-gamma hand and foot monitors, alpha probe counters, beta-gamma probe counters, are also required.

UNLESS OTHERWISE SPECIFIED USE THE FOLLOWING:--			
APPLIED PRACTICES	SURFACES	REVISIONS OR REVISIONS	REVISIONS
	✓	+	+

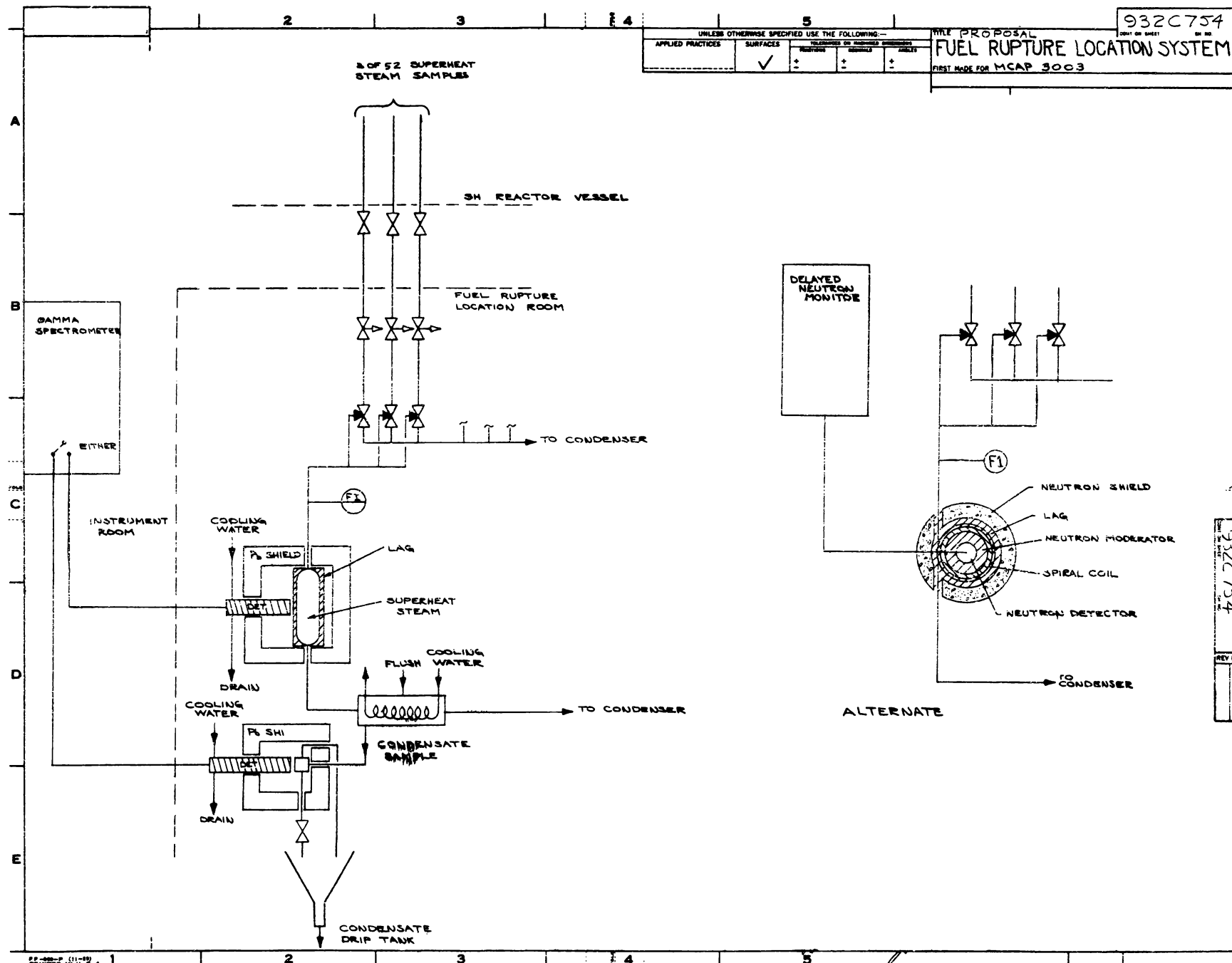
FILE PROPOSAL
FUEL RUPTURE LOCATION SYSTEM
 FIRST MADE FOR MCAP 3003

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DATE OF SHEET

REV NO

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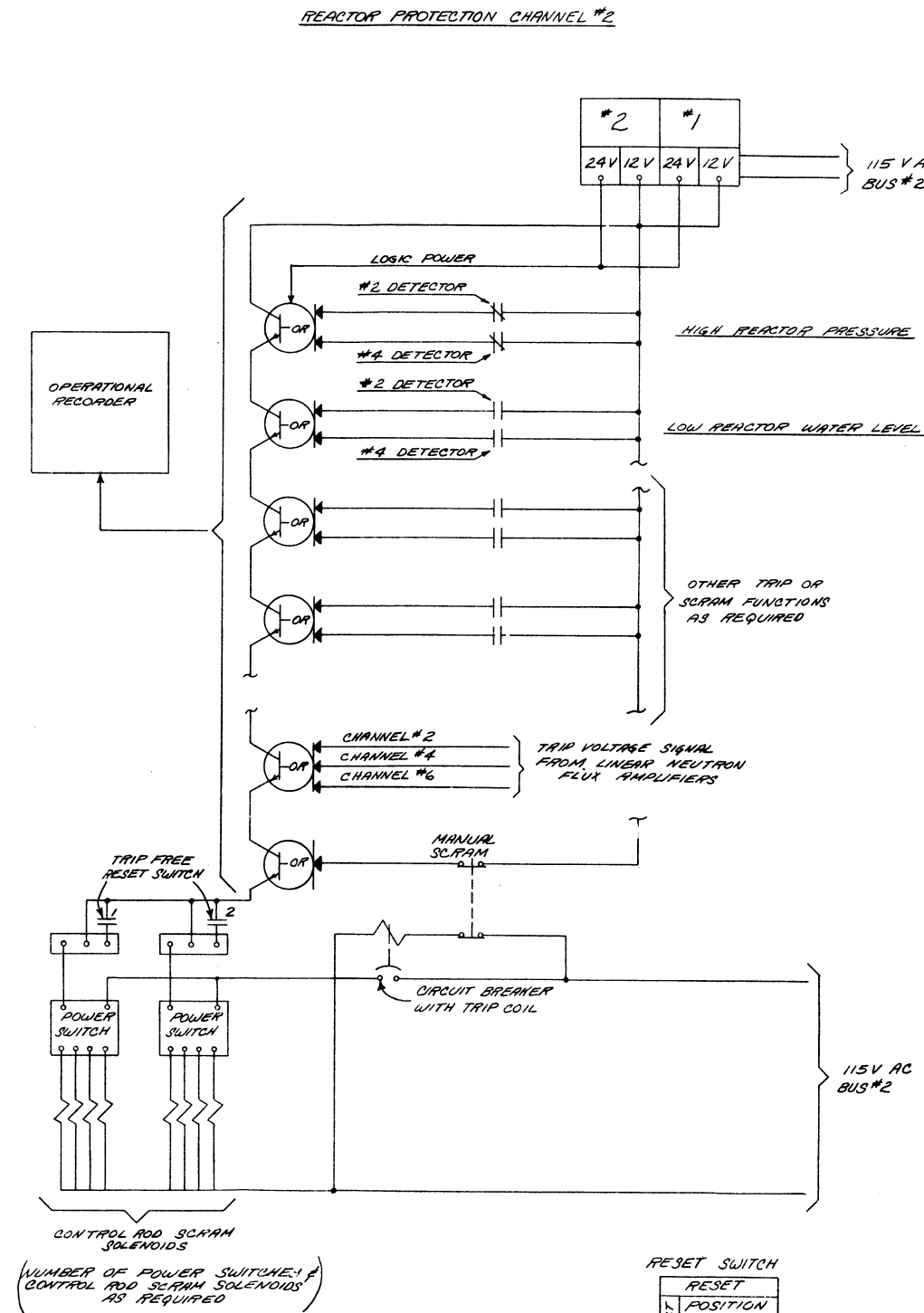
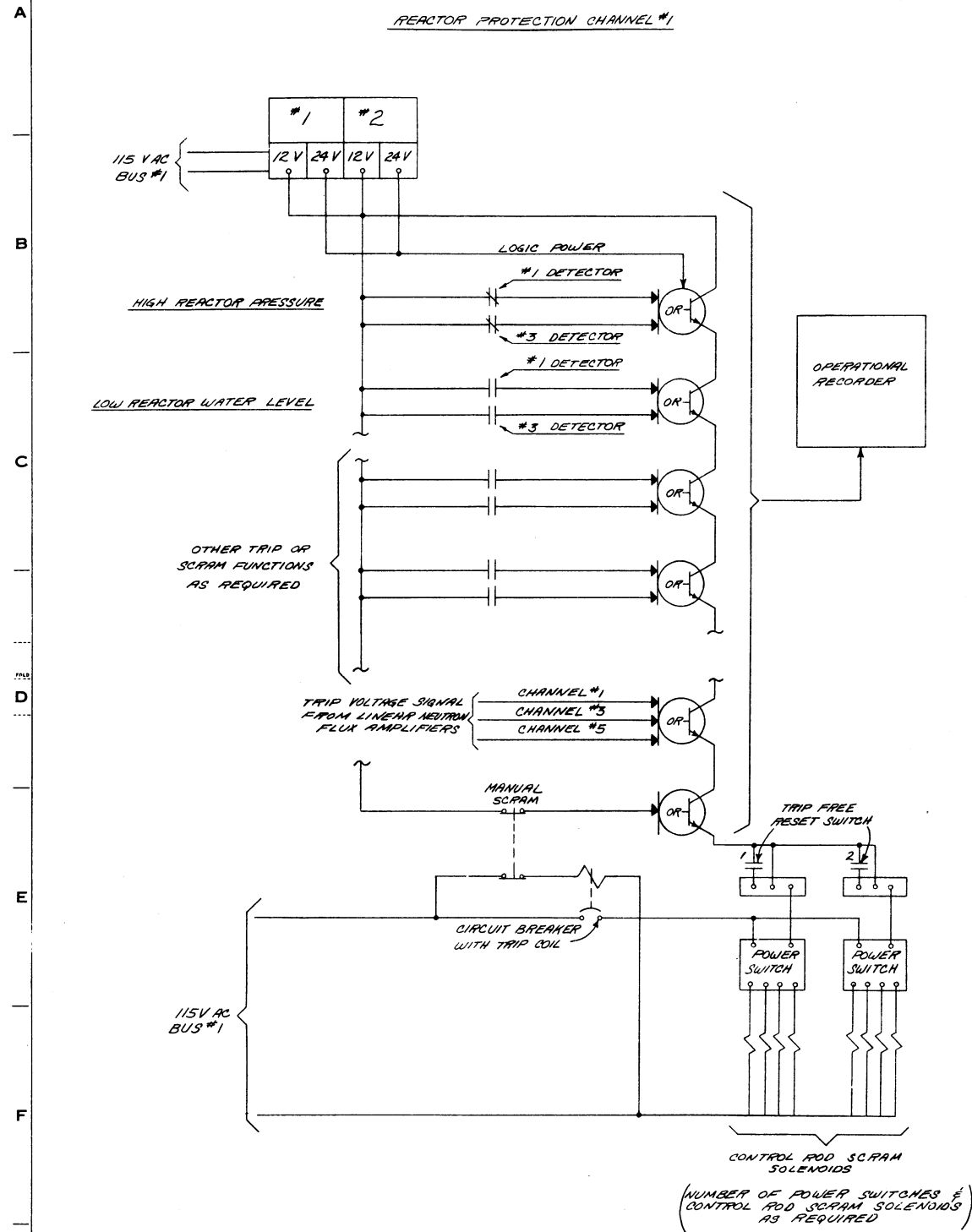
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DEFINITION OF LOGIC ELEMENTS

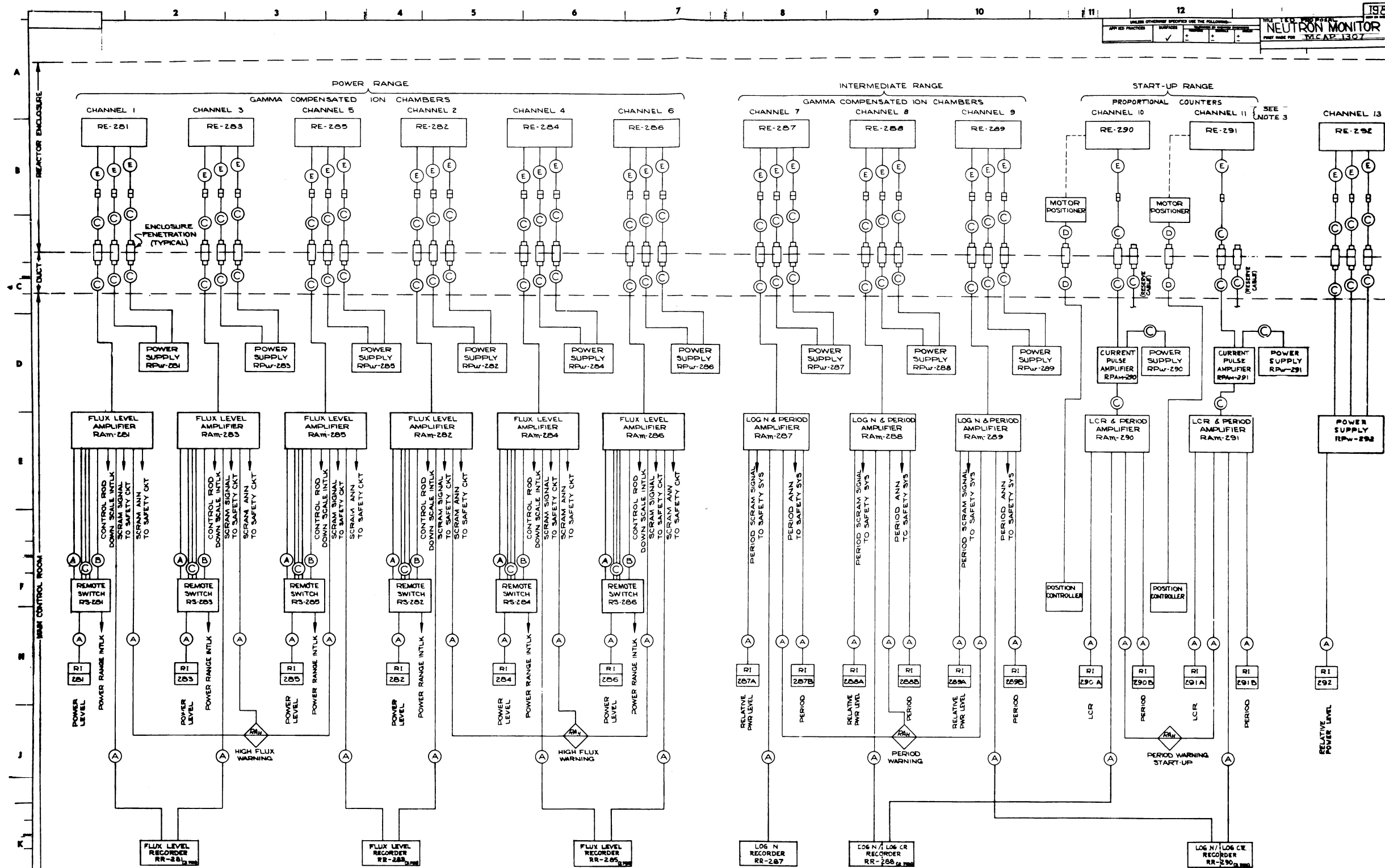
TRIP* SIGNAL $\left\{ \begin{array}{l} \text{INPUT 1} \\ \text{INPUT 2} \end{array} \right\} \text{OR} \text{TRIP OUTPUT SIGNAL*}$
 *OR LOGIC - A TRIP OUTPUT SIGNAL IS DERIVED FROM THE PRESENCE OF A TRIP AT EITHER INPUT OR AT BOTH INPUTS.

*TRIP SIGNAL IS INDICATED BY ZERO VOLTAGE



RESET SWITCH

CONTACT	POSITION	1	2
1	RESET	X	
2	RESET		X



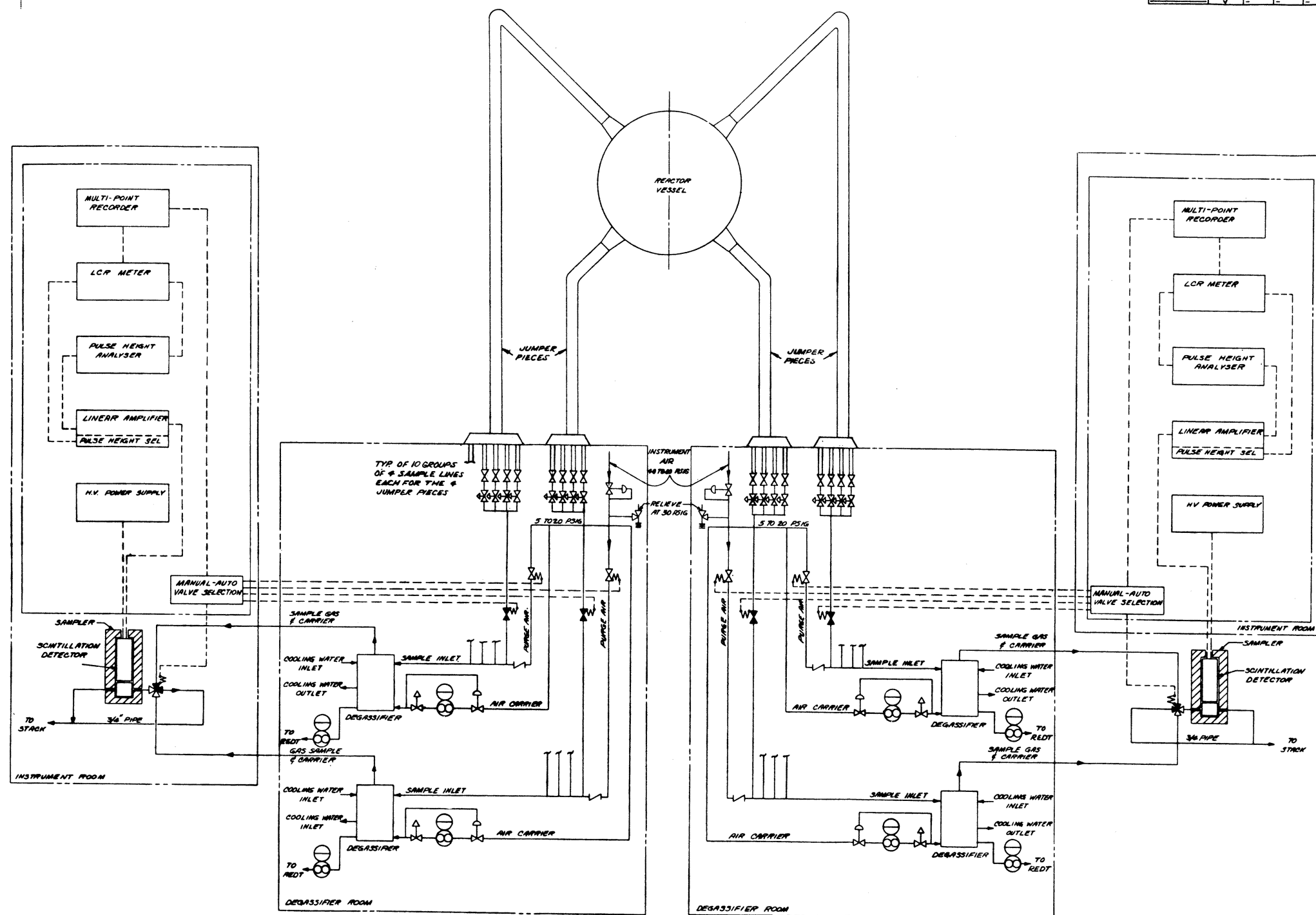
CABLE LEGEND

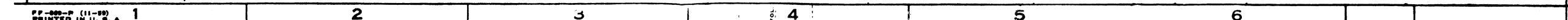
SYMBOL—○—ON DWG INDICATES CABLE PER CHART BELOW. THESE CABLES ARE NOT SUPPLIED BY APED UNLESS OTHERWISE SPECIFIED.

SYM	DESCRIPTION
A	2 COND. #18 AWG WITH OVERALL SHIELD, EQUAL TO GE TYPE 34-50-000
B	2 COND. #18 AWG EQUAL TO GE TYPE 34-50-000
C	NOISE-FREE RG-59/U, EQUAL TO ANTIMICROPHONIC TYPE MADE BY SIMPLEX CABLE & WIRE CO
D	#16 COND. #14 CABLE
E	SPECIAL HIGH TEMP RADIATION RESISTANT CABLE, RG-59/U, NOISE FREE, APED SUPPLIED

GENERAL NOTES:

- 1- POWER RANGE CHANNELS 1 THRU 6 CAUSE ALARM INDICATION ON EXCESS NEUTRON FLUX AND INITIATE SCRAM WHEN ONE ODD- AND ONE EVEN NUMBERED CHANNEL INDICATES FLUX SCRAM LEVEL.
- 2- INTERMEDIATE RANGE CHANNELS 7, 8 & 9 CAUSE ALARM INDICATION ON APPROACH TO SHORT PERIOD AND INITIATE SCRAM WHEN 2 OF THE 3 CHANNELS INDICATE REACTOR IS ON SHORT PERIOD. THESE CHANNELS ARE BY-PASSED IN THE POWER RANGE.
- 3- START-UP RANGE CHANNELS 10, 11 & 12 CAUSE ALARM INDICATION ON SHORT PERIOD. CHANNEL 12 SIMILAR TO CHANNEL 10 (11 IS NOT SHOWN).
- 4- THE SIGNAL TO BE RECORDED IS SELECTED BY A MANUAL SWITCH IN THE RECORDER.
- 5- CHANNEL 13 IS DC POWERED AND MONITORS THE FLUX DECREASE FOLLOWING THE SCRAM WHEN AUXILIARY POWER IS LOST.



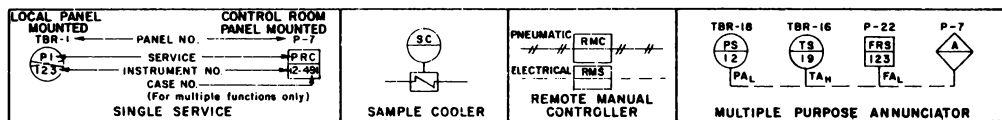


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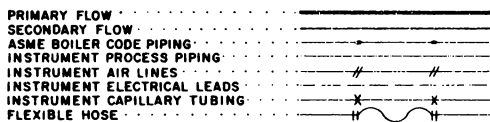
INSTRUMENT SYMBOLS

INSTRUMENT FUNCTION		CONTROLLING				MEASURING										SWITCH		ALARMS					POWER SUPPLY
		RECORDING	INDICATING	BLIND	CONTROL VALVE	RECORDING	INDICATING	OBSERVATION GLASS	PRIMARY ELEMENT	TEST POINT	TRANSMITTER	INTEGRATOR	AMPLIFIER	SAMPLER	INDICATING	BLIND	ALARM	ALARM LOW	ALARM HIGH				
MEASURED VARIABLE		-RC	-IC	-C	-CV	-R	-I	-G	-E	-X	-T	-Q	A _m	S _m	-IS	-S	-A	A _L	A _H				
AIR	A													AS _m									
CONDUCTIVITY	C	CRC	CIC		CCV	CR	CI		CE	CX				CS _m	CIS	CS	CA	CA _L	CA _H				
DENSITY	D	DRC	DIC	DC	DCV	DR	DI			DX					DIS	DS	DA	DA _L	DA _H				
DIFF. PRESS	dP	dPRC	dPIC	dPC	dPCV	dPR	dPI				dPT				dPIS	dPS	dPA	dPA _L	dPA _H				
FLOW	F	FRC	FIC	FC	FCV	FR	FI	FG	FE	FX	FT				FIS	FS	FA	FA _L	FA _H				
HUMIDITY	H	HRC	HIC			HR	HI		HE														
pH	pH	pHRC	pHIC	pHC	pHCV	pHR	pHI		pHE	pHX			pHA _m	pHS _m									
LEVEL	L	LRC	LIC	LC	LCV	LR	LI	LG			LT				LIS	LS	LA	LA _L	LA _H				
OXYGEN	O ₂					O ₂ R	O ₂ I		O ₂ E														
PRESSURE	P	PRC	PIC	PC	PCV	PR	PI			PX	PT				PIS	PS	PA	PA _L	PA _H				
POSITION	Po					PoR	PoI			PoT					PoS								
RADIATION	R					RR	RI		RE	RX			RA _m	RS _m		RS	RA			RP _w			
SPECIFIC GRAVITY	Sg																						
SPEED	Sp	SpRC	SpIC	SpC		SpR	SpI		SpE														
TEMPERATURE	T	TRC	TIC	TC	TCV	TR	TI		TE	TX	TT				TIS	TS	TA	TA	TA				
TURBIDITY	Tu																						
VISCOSITY	V																						
VIBRATION	Vb															VbS	VbA						
WEIGHT	W																						
WEIGHT FACTOR	Wf	WfRC	WfIC		WfV																		
WATER	Wa														Suffix sub H or L may be added to designate High or Low.								

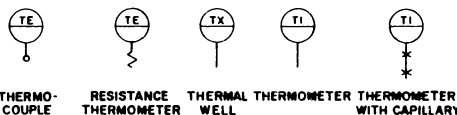
TYPICAL INSTRUMENT SYMBOLS



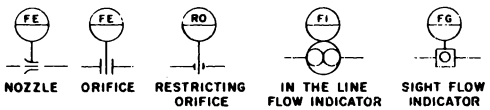
LINE SYMBOLS



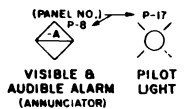
TEMPERATURE ELEMENTS



FLOW DEVICES



ALARMS



TRANSMITTERS

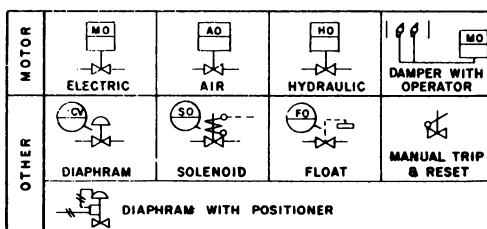


MISCELLANEOUS

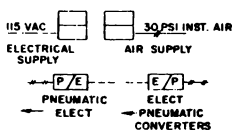


- NOTES: 1. FIRST LETTER OF INSTRUMENT SYMBOL SHALL BE PROCESS VARIABLE.
2. A SWITCH IS A DEVICE CAPABLE OF OFF - ON ACTION.
3. AN ALARM IS THE ANNUNCIATOR, SIGNAL HORN OR OTHER DEVICE, ACTUATED BY SWITCH CONTACTS.

OPERATORS



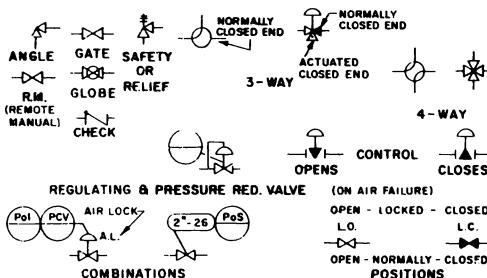
INSTRUMENT POWER SUPPLY



INSTRUMENT AIR SET



VALVES



6.0 Service System

6.1 Closed Cooling Water System

Certain components of the nuclear steam supply system require a closed loop cooling system to guard against the possibility of a gross distribution of radioactive water to the plant service water system. These components are:

- Cleanup demineralizer non-regenerative heat exchanger
- Recirculating pump cooling system
- Radioactive enclosure drain tank
- Biological shield cooler
- Cleanup system recirculation pump cooler
- Sample coolers

Water is pumped in a closed system to each of the above systems. The heat absorbed is then transferred to station service water in the closed cooling system heat exchanger. Normal design practices will govern the design of this equipment. It is recommended that two full size pumps and heat exchangers be provided, and that a surge tank be included to protect the system from normal operating transients.

6.2 Service Water System

A conventional station service water system should be provided to remove heat generated in the various equipment (i.e., station air compressors). In addition to those normal requirements, certain special systems unique to the nuclear plant will require service water cooling. These items are:

- Shutdown heat exchangers.
- Shutdown pump coolers
- Dry well cooler
- Water vapor condenser (waste system)
- Waste collector tank cooler (waste system)
- Suppression pool (reactor containment system)

Service water may be safely used in the above systems since they are low level radioactivity mediums and/or equipment used only during reactor shutdown periods.

6.3 Liquid Waste System

6.3.1 System Description and Criteria

Many of the wastes are intermittent or irregular in nature depending on the frequency of refueling and the need for maintenance. Therefore, the waste system is sized for peak loads as well as for daily loads.

In general, the wastes are ultimately collected in either the waste collector tank, in the floor drain collection tank, or in the waste neutralizer tank. These also serve as feed tanks for subsequent waste treatment equipment -- the demineralizer, the filter, or waste concentrator.

The waste system is designed so a minimum of wastes are discharged to the river. Most wastes are processed such that the treated water is suitable for return and reuse in the plant as make-up.

Waste resins or concentrated wastes are collected in interim storage tanks where they are held for shipment to off-site disposal sites. Although several alternate methods of off-site shipment and disposal are available, that chosen for study use was the use of a waste disposal contractor. On this basis concentrated wastes would be put into 3000-4000 gallon tank trucks for hauling away. Spent resins would be put into 55-gallon drums for off-site disposal. Therefore, the concentrated waste receiver is sized to hold 8000 gallons to give some flexibility and freedom from truck schedules. The spent resin receiver was sized to receive two batches of resin to again permit flexibility in time of loadout into drums.

6.3.2 Flow Diagram and Data

The waste disposal flow diagram (Drawing No. 198E147) shows all the waste routings, tanks, equipment, etc. Sizes, numbers and types of tanks and equipment are shown. The following tabulations also give some of this information.

<u>Collection Tanks</u>	<u>Capacity of Size</u>	<u>Material</u>
Chemical Addition Tank	300 Gal.	Rubber-lined steel
Waste Collector	25,000 Gal.	Carbon Steel
Reactor Enclosure Drain Tank	5,000 Gal.	Carbon Steel
Waste Neutralizer	12,000 Gal.	Plastic-lined steel
Floor Drain Collection Tank	10,000 Gal.	Carbon Steel

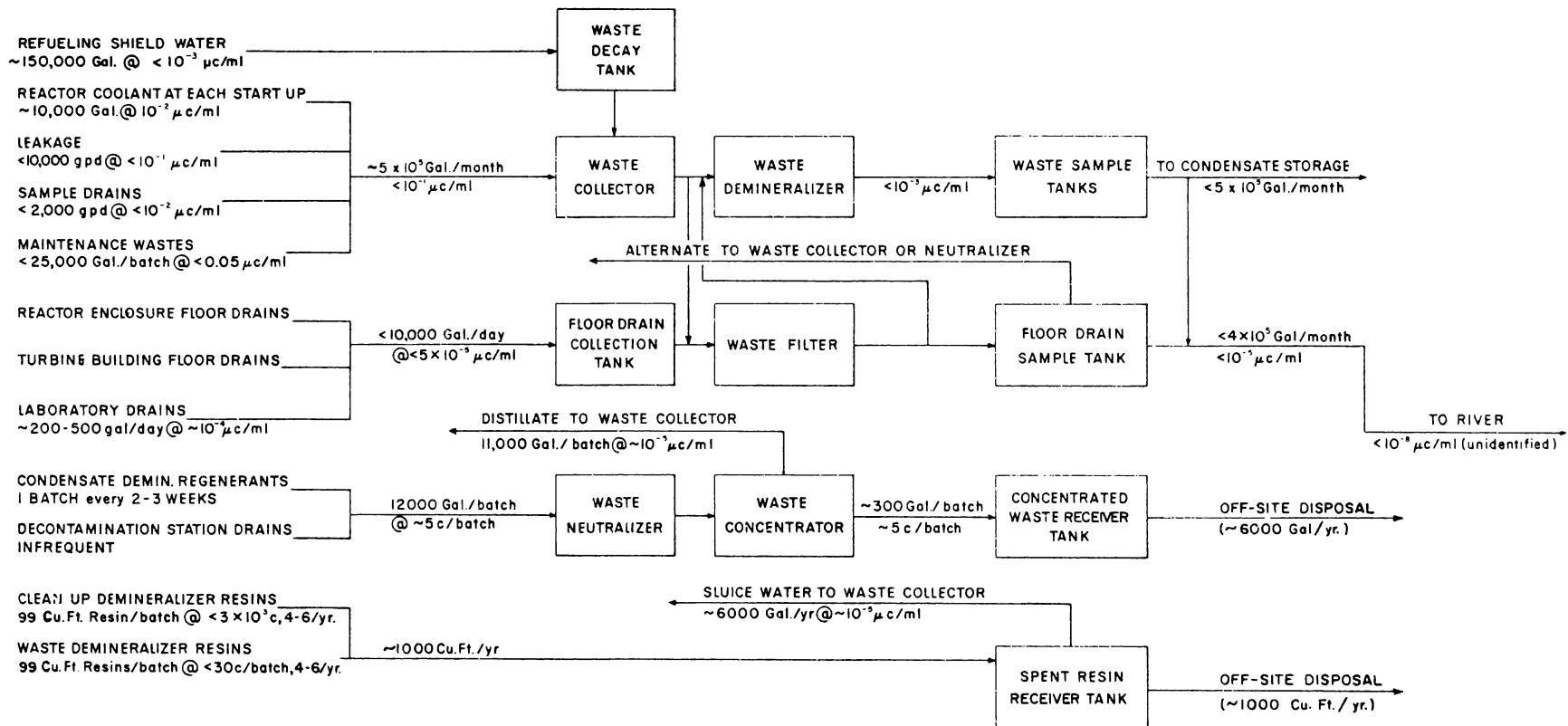
Treatment Equipment

Waste Filter	250 gpm	Carbon Steel
Waste Demineralizer	250 gpm	Plastic-lined steel
Waste Concentrator Including Vapor De-entrainer	150 gal/hr	Stainless Steel

Retention and/or Storage

Waste Sample Tanks	2 @ 25,000 Gal. Ea.	Aluminum
Floor Drain Sample Tank	10,000 Gal.	Carbon Steel
Concentrated Waste Storage	8,000 Gal.	Carbon Steel
Resin Storage Tank	2,000 Gal.	Stainless Steel
		AISI - 316
Waste Decay Tank	150,000 Gal.	Carbon Steel

As mentioned in the above, treatment is either by filter or demineralizer for removal of radioactivity or by concentration to a small volume (20-30% solids by weight) for subsequent storage.



ESTIMATED QUANTITIES & ACTIVITIES OF RADIOACTIVE LIQUID WASTES

The demineralizer produces decontamination factors of 10^2 minimum, and effluent activities of 10^{-3} uc/ml or less are expected depending upon influent activity. This is adequate for wastes to be returned to the plant. Whenever waste activities are low enough (10^{-4} uc/ml) and a large enough source of dilution water is available, e.g., condenser cooling water, wastes may be diluted to isotopic concentrations less than the mps and such wastes may then be safely discharged to the river.

The vapor de-entrainer above the concentrator is designed for a maximum solids content of the condensate of 2 ppm. Further decontamination of this is accomplished by the demineralizer.

Large decay times have not been specifically provided since many of the activated corrosion products have half-lives of 30 or more days. Decay of these would not prove very fruitful, as a method of daily processing. The significant short half-lived corrosion products are Mn-56 and Ni-65 with half-lives of about 2-1/2 hours and 12 hr. Cu-64. Decay time of several days will reduce the activity of these contributions to about that of the longer lived radioisotopes. Such decay times can be obtained by using the waste decay tank.

6.3.3 Flexibility

As may be noted by inspection of the flow diagram, the system permits considerable operational flexibility. First, tank and treatment equipment capacity is based on peak loads expected as well as daily loads. Since the large quantities of wastes are due to condensate demineralizer regeneration and maintenance operations, ample capacity is available for processing the daily wastes. Second, wastes received in the waste collector, floor drain collection tank and waste neutralizer may be treated by either filtration and demineralization or concentration. Also, if of low enough activity, they may be discharged off-site. Third, off-standard wastes may be returned for retreatment or concentration and storage. Fourth, treated wastes may be returned to the plant for reuse or discharged to the river, provided discharge specifications are met. Fifth, adequate drain lines are provided so that all plant equipment and piping may be flushed and drained to the waste system as necessary for maintenance.

6.3.4 Types of Construction

To prevent leakage to ground and subsequent contamination of ground water, secondary containment is provided for radioactive waste piping and equipment. Means of detection of leaks or overflows are also provided. The containment is provided by locating the tanks and equipment within the buildings, dikes or concrete vaults. Piping outside of buildings is encased in concrete trenches. Exceptions are lines carrying treated wastes having low enough activity so significant ground contamination would not occur. Sumps with liquid level devices are provided to detect leaks.

Tanks are of welded construction and materials are suitable for the services. Piping is primarily welded although some flanged connections are used. No screwed fittings are employed.

6.3.5 Other Features

Whenever wastes are discharged to the river, a sampler is located on the discharge canal to provide a composite sample of water discharged to the river. This serves as a check on waste system discharges.

Further, an environs monitoring program is recommended both prior to operation and after operation to provide data on the effect (if any) of plant operation on environmental radioactivity.

6.3.6 Control

Liquid waste system operation is controlled from a local panel in the waste building. Operation is remote rather than at the equipment because of radiation and the location of most of the equipment behind shielding.

6.4 Solid Wastes

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

6.4.1 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.

6.4.2 Methods of Handling and Disposal

The general plan is to temporarily hold the solid wastes on the plant site in a separate controlled area until suitable amounts are on hand for shipment to a permanent off-site disposal site. However, used reactor equipment is 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.

Wastes received in the waste collector are generally demineralized water with varying amounts of radioactivity depending upon from which portion of the plants the wastes derive. Activity of these wastes range from about one to 10^{-5} microcuries per milliliter (uc/ml) depending upon the source and proportion of fission and corrosion products. Usually these wastes are treated by deionization, then sampled and returned to condensate storage for reuse in the system. Alternately, if of low enough activity, or if they can be adequately diluted, they can be discharged to the river.

Wastes received in the floor drain collection tank are primarily floor drains. These wastes are usually low in activity (normally less than 5×10^{-5} uc/ml) but they may be high in solids and dirt content. Thus, they may be most easily disposed of to the river with dilution as required. Filtration is provided to minimize particle discharge to the river and to permit demineralization as an alternate treatment.

Wastes received in the waste neutralizer are usually corrosion wastes or wastes comparatively high in dissolved solids content. They may also contain high concentrations of activity. For these reasons the most suitable treatment is reduce their volume as much as possible by evaporation and store the concentrate indefinitely. To make the equipment materials following the neutralizer as economical as possible, the wastes are made slightly basic (pH = 7 - 9) as necessary with the addition of a caustic.

The waste system is designed as a batch system instead of as a continuous type of system because many of the wastes are of irregular frequency and especially because it is necessary to maintain control of all discharges from the system. Thus, the suitability of each batch of waste for disposal must be first determined by sampling and laboratory analysis.

Wastes to be returned to the plant must be low enough in activity so radiation levels in storage tanks, piping or equipment are not significantly affected. An activity concentration of about 10^{-3} uc/ml meets this requirement. Also, these wastes must meet feed-water purity requirements - pH 7.0 - 7.2 and conductivity of 1 micromhos.

Wastes discharged to the river should have radioisotope concentrations at the plant boundary and after any dilution which are below the maximum permissible concentrations (mpc) as stipulated in Title 10 - Part 20, "AEC Standards for Protection Against Radiation." For unidentified mixtures, this mpc is 10^{-7} uc/ml of beta-gamma and of alpha activity, since radium isotopes will not be present. For mixtures containing known radioisotopic concentrations (as by analysis) the mpc may vary from a minimum of a total beta-gamma or alpha count to more complete analyses for radioisotopes having low mpc's to establish the mpc for the batch. Treated wastes meeting neither of the above disposal requirements are returned to the waste system for reprocessing.

Most of the solid wastes are not so radioactive as to preclude handling by contact. They should 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) are 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). The containers are then sealed and moved to a central collection area.

Where possible, wastes which are compressible are compressed in a conventional press and baling machine to reduce their volume. The press, which exerts a pressure of about 16,000 psi and reduces the volume to about $\frac{1}{4}$ of the original volume, is located in a small building in the storage yard. Filtered ventilation air flowing past the press prevents spread of particulate contamination which may result from compression of solid wastes.

Compressed or not, the packaged solid wastes are periodically shipped in approved containers to a permanent disposal site.

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

6.5 Off-Gas System

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 hydrogen and oxygen.

6.5.1 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 provide 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. Minimize the safety hazard due to the presence of hydrogen and oxygen in explosive concentrations.

6.5.2 Main Condenser - Air Ejector Off-Gas System

This system provides a 30 minute holdup of gases prior to release from the stack. This includes the minimum decay time of five minutes required to allow decay of N^{16} and O^{19} to negligible radiation levels. The remaining hold-up time permits the operator to take suitable action to reduce or otherwise control the release of noble gases should they be generated at high rates.

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.

Flowmeters to measure off-gas flow, and temperature and pressure recorders are provided. A valve near the stack will permit closing the system to retain the gases if necessary. A filter near the stack is provided to remove particulate matter.

The system is designed to contain explosions, except at the air ejector after-condenser. Here, a rupture disc is provided to protect against over-pressurization. A steam system is provided which will automatically dilute the gases to less than 4% hydrogen (the lower explosive limit) on a system pressure or temperature rise. The steam system capacity provides for at least five minutes dilution.

6.5.3 Gland Seal Condenser Off-Gas System

This off-gas system provides 90-second holdup between the gland seal condenser and the stack for decay of N^{16} to a negligible radiation level. Provision is made for future addition of a preheater and charcoal filter to this system should they prove necessary. However, only a small fraction of the fission gases follow this route so this is only a precaution. Explosive mixtures are not present. The system is kept separate from the air ejector system. Gases removed from the condenser by the vacuum pump at startup should be routed to the stack via the gland seal off-gas line.

6.5.4 Ventilation and Vents Off-Gas System

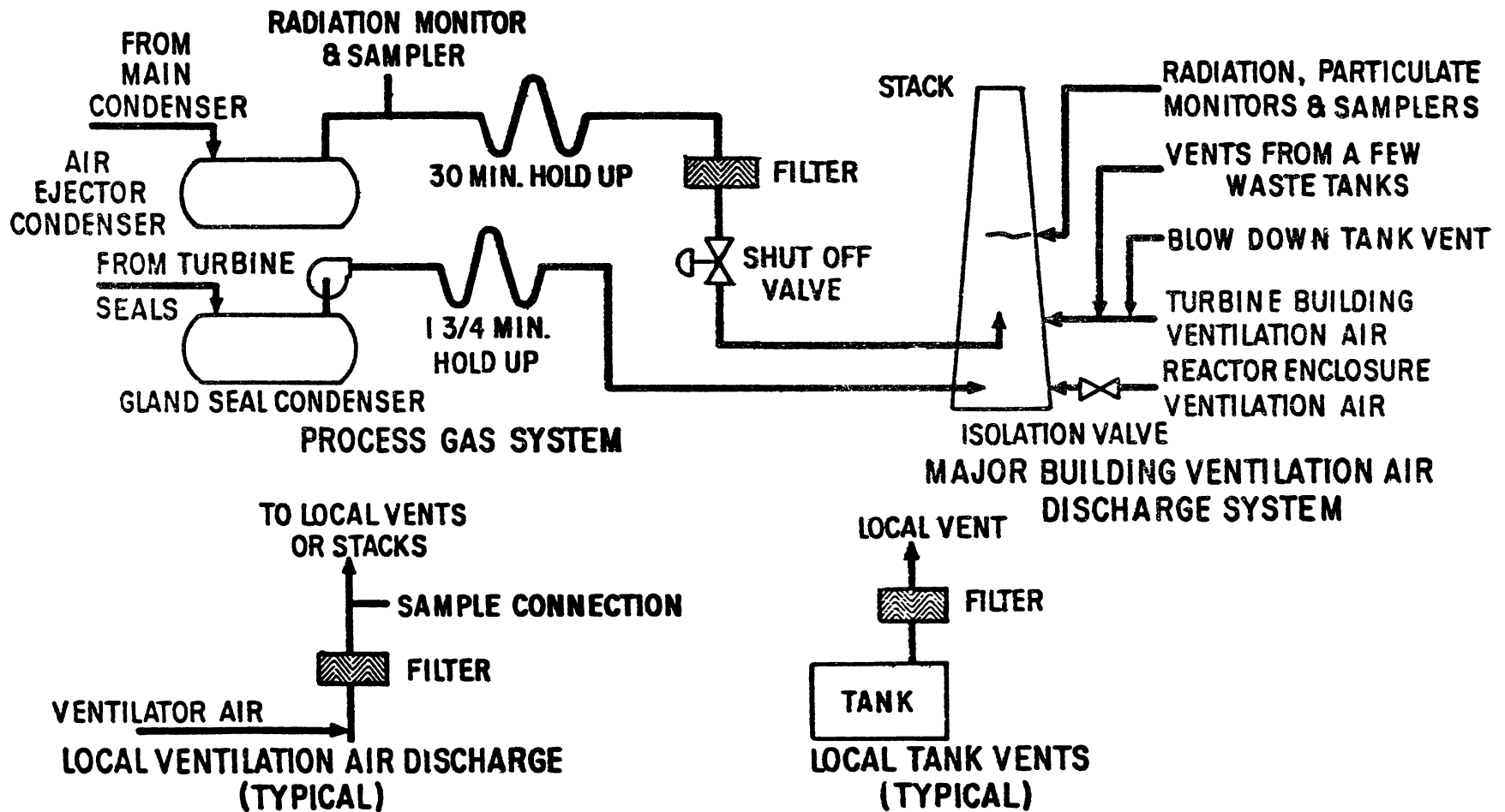
The other source of gaseous radioactive waste is the ventilation air and gaseous vent discharge from processing equipment or storage tanks for radioactive material. The reactor enclosure and turbine building ventilation air contributes the greatest volume of effluent air from the plant. The potential for the presence of significant particulate matter in ventilation air from these sources is small, and therefore this air is not filtered. It is routed to the main stack, however, to provide for dispersal in the remote event of a process leak. An additional reason for discharge of ventilation air from the reactor enclosure to the stack is that it occasionally includes air from around the reactor vessel. This air contains Argon-41, which is a radioactive gas,

not removable by filtration. The air around the reactor vessel is released to the stack after suitable decay only before refueling.

Radioactive particulate matter is the primary source of radioactivity in gases or air from vents. The radioactivity is removed from these gases by use of high-efficiency filters. In addition, these gases are discharged through the main stack or through small stacks or vents to provide some dilution and dispersion into the atmosphere for the small quantities of activity which may remain. The continued effectiveness of these measures is observed by periodic gas sampling and area surveys using both permanent and portable instruments. Any unusual release from these sources generally results in a plant site problem rather than an environs problem so corrective action, if needed, can be taken before any significant environs exposure results.

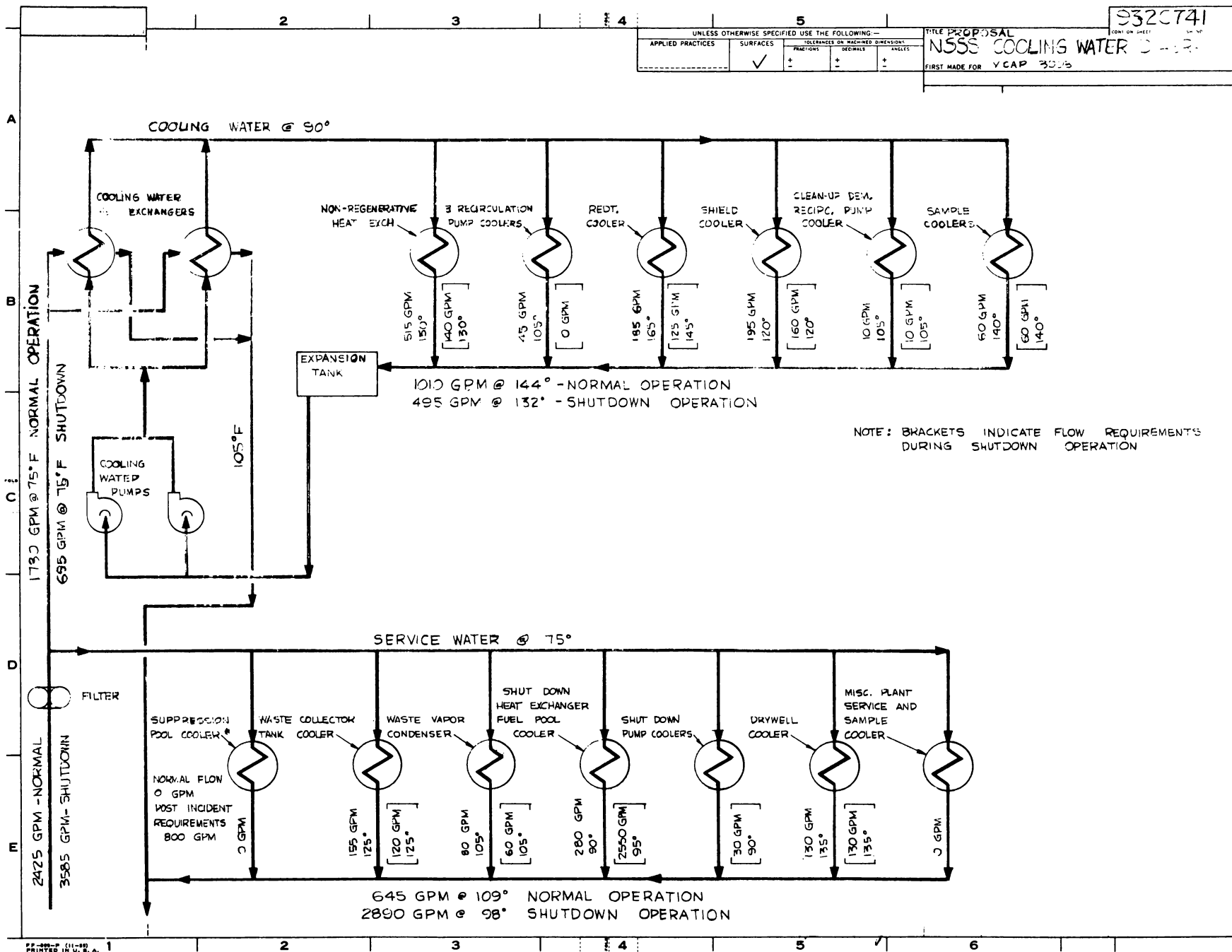
Vents and ventilation air from equipment or buildings containing non-radioactive material are usually kept separate from gases from radioactive zones or sources, and are discharged to the atmosphere in a conventional manner.

SCHEMATIC DIAGRAM OFF GAS DISPOSAL SYSTEM



UNLESS OTHERWISE SPECIFIED USE THE FOLLOWING:-			
APPLIED PRACTICES	SURFACES	TOLERANCES ON MACHINED DIMENSIONS	
✓	✓	FRACTIONS	DECIMALS
			ANGLES

9320741
CONT. ON SHEET
TITLE PROPOSAL
N555 COOLING WATER SYSTEM
FIRST MADE FOR VCAP 30/5





1. ANY WASTES DISCHARGED TO THE RIVER SHALL BE AT A CONCENTRATION OF 0.1 MPC FOR THE MIXTURE AS IT ENTERS THE RIVER. MPC VALUES ARE ABOVE BACKGROUND OF ACTIVITY NORMALLY OCCURRING IN RIVER WATER.
2. MPC = MAXIMUM PERMISSABLE CONCENTRATION
3. ALL TANK AND EQUIPMENT VENTS SHALL BE FILTERED TO REMOVE ANY RADIOACTIVE PARTICLES BEFORE DISCHARGE TO THE ATMOSPHERE.
4. ALL TANKS SHALL HAVE OVERFLOWS
5. ALL TANKS SHALL HAVE HIGH LEVEL ALARMS & LOW LEVEL SHUT OFF OR DISCHARGE PUMPS

REF. DWG.

11485043 FOR TYPICAL WASTE FILTER
PIPING INCLUDING FILTER AID
ADDITION & BACKWASHING.

7.0 Reactor Refueling and Maintenance

Reactor servicing is separated into two categories: refueling and maintenance. During refueling, preliminary servicing is necessary to provide access to the reactor core. This consists mainly of removing the dry well head shielding, reactor head, and internals over the core. After initial preparation has been made, the refueling procedure follows.

Reactor maintenance will be necessary at less frequent intervals during the life of the reactor. Maintenance will be scheduled so that it can be accomplished during the refueling shutdowns to minimize plant downtime. All reactor core components are designed for ready removal and replacement when necessary. Reactor refueling and servicing are somewhat interdependent, but the flexibility built into the system allows for variation in sequential operations, and permits simultaneous refueling operations, decreasing plant shutdown costs.

7.1 Refueling

7.1.1 Fuel Handling

The major components supplied for fuel handling have been evaluated to provide an overall cost improvement in the system while maintaining the features of inherent safety, flexibility, and ease of operation and maintenance. The refueling arrangement adopted minimizes fuel handling and other time-consuming operations. Water is used to shield the operators and permit direct observation of underwater operations. Refueling operations are carried on simultaneously above the reactor and over the fuel storage pool from moving bridges. A large-diameter transfer tube provided with a gate valve connects the reactor well with the fuel storage pool. The valve is closed during reactor operation, and opened during the refueling operation after the reactor well has been flooded. The carrier, operating in the transfer tube, is used to move the spent fuel from the reactor well to the fuel storage pool. The transfer carrier is powered independently, leaving the hoists and bridges free for other fuel handling operations while the fuel is being transported. The pool water provides an adequate heat sink for the decay heat from the fuel assembly.

Simple handling tools are used to grapple fuel in the core and to load the transfer carrier. A duplicate set of tools is used in the storage pool to grapple fuel in the carrier and transfer it to storage racks. These tools, evolved from past experience at Dresden and other operating plants, are used in conjunction with a hoist mounted on each operating bridge.

A similar refueling procedure is planned for the superheat reactor. Space is provided in the pool for decay storage of superheat fuel. Pool heat removal and water cleanup equipment is sized for combined reactor fuel storage requirements. An additional bridge, transfer tube, and carrier are required to service the superheat reactor, along with different fuel grapples and other hand tools. To reduce

the length of the superheat fuel assembly to be handled, the steam inlet lines are removed and stored temporarily in the reactor well. The active fuel is handled like the boiling reactor fuel. The steam inlet tubes are reused on new fuel assemblies.

7.1.2 Fuel Storage

Controlled conditions are required for storage of new and spent fuel to prevent assembly of a critical array outside of the reactor, to shield the operators from spent fuel, and for accountability of nuclear material assigned to the plant. A storage pool water cooling system is required to remove decay heat from spent fuel.

New fuel requires no shielding, but storage spacing must be adequate to avoid criticality in case the vault is flooded. Normally, enough fuel is stored on site to cover refueling requirements, plus a few replacements for elements that might be damaged in handling. It is also desirable to provide space in or near the vault for inspection of incoming fuel assemblies. Temporary storage can be considered for the bulk of the first core loading. Use of the fuel racks in the storage pool is one possibility. Provision is required for mechanized handling of fuel from the vault to the reactor area.

Spent fuel must be stored under water to shield the operators and for removal of decay heat until fuel activity level has decayed enough to permit economical shipment for reprocessing. Poisoned fuel racks are required to minimize storage pool size, and therefore, the cost of the surrounding building.

Storage space for 120% of a core loading is required so that the entire core may be unloaded if necessary while a batch equal to 20% of the core loading is stored for decay prior to shipment for reprocessing.

Shipment of fuel for reprocessing involves loading the shipping cask with spent fuel in the storage pool, decontamination of the cask exterior, and placement of the cask on a railroad car at the site. Pool space is provided for loading the shipping cask which must be handled with the large building crane.

7.2 Maintenance

7.2.1 Channel Handling

The fuel channels (used in the boiling water reactor only) are designed to remain with the core structure, and only the fuel is removed during refueling. This arrangement has proven satisfactory for the Vallecitos Boiling Water Reactor, and allows a more economic system for reuse of channels. In the event a channel is damaged, it may be readily removed and replaced with another. The channels are connected to the bottom grid of the core structure by spring clips which are released by the channel handling tool. The channel is then withdrawn, placed in the transfer carriage, and moved to the pool for storage.

7.2.2 Control Rod Handling

Control rod life approximates core life but depends on the time the control rod spends in high flux regions of the core. Control rod strength is normally checked during critical testing after each change in core loading. When replacement is indicated, the fuel assemblies and the channels surrounding a control rod are removed. The control rods are attached to the drive mechanisms 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 via the transfer tube and carrier.

Irradiated control rods must be shielded, but the depth of water required is less than for fuel.

Although the control rods for the superheat reactor are box-shaped rather than cruciform, they are coupled to the offset drives with a similar coupling that can be released from above. A stop at the top of the core support and control rod guide structure must be removed from above before the control rod can be lifted out of the core.

7.2.3 Control Rod Drive Mechanism Removal and Replacement

Infrequent drive maintenance is required to inspect wear surfaces and replace seals and other worn parts. This maintenance 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 by using an activating cylinder and external jack screws built into the drive. Water in the drive is drained, drive mounting bolts are removed, and the drive is lowered to a handling dolly using 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 using essentially the same procedure described above. The control rod is removed by uncoupling from above. A thimble is lowered over the control rod, and seals against the control rod drive thimble at the bottom of the reactor vessel. The drive is then drained and lowered in the conventional manner. Partial drive disassembly may be required to completely remove the drive from the thimble.

7.2.4 In-Core Flux Monitor Servicing

The in-core flux monitors are positioned in guide tubes located between channels, and need not be disturbed during normal refueling. In order to replace a string of flux monitors, the electrical connections below the reactor are broken, a holddown fitting in the

flux monitor seal flange is removed and replaced with a thimble; then the flux monitor assembly is withdrawn from above. Water in the well above the reactor shields the operators from the radioactive portion of the flux monitor assembly. A new assembly is threaded down through the core from above using a sleeve to stiffen, guide, and protect the connectors on their way down through the core.

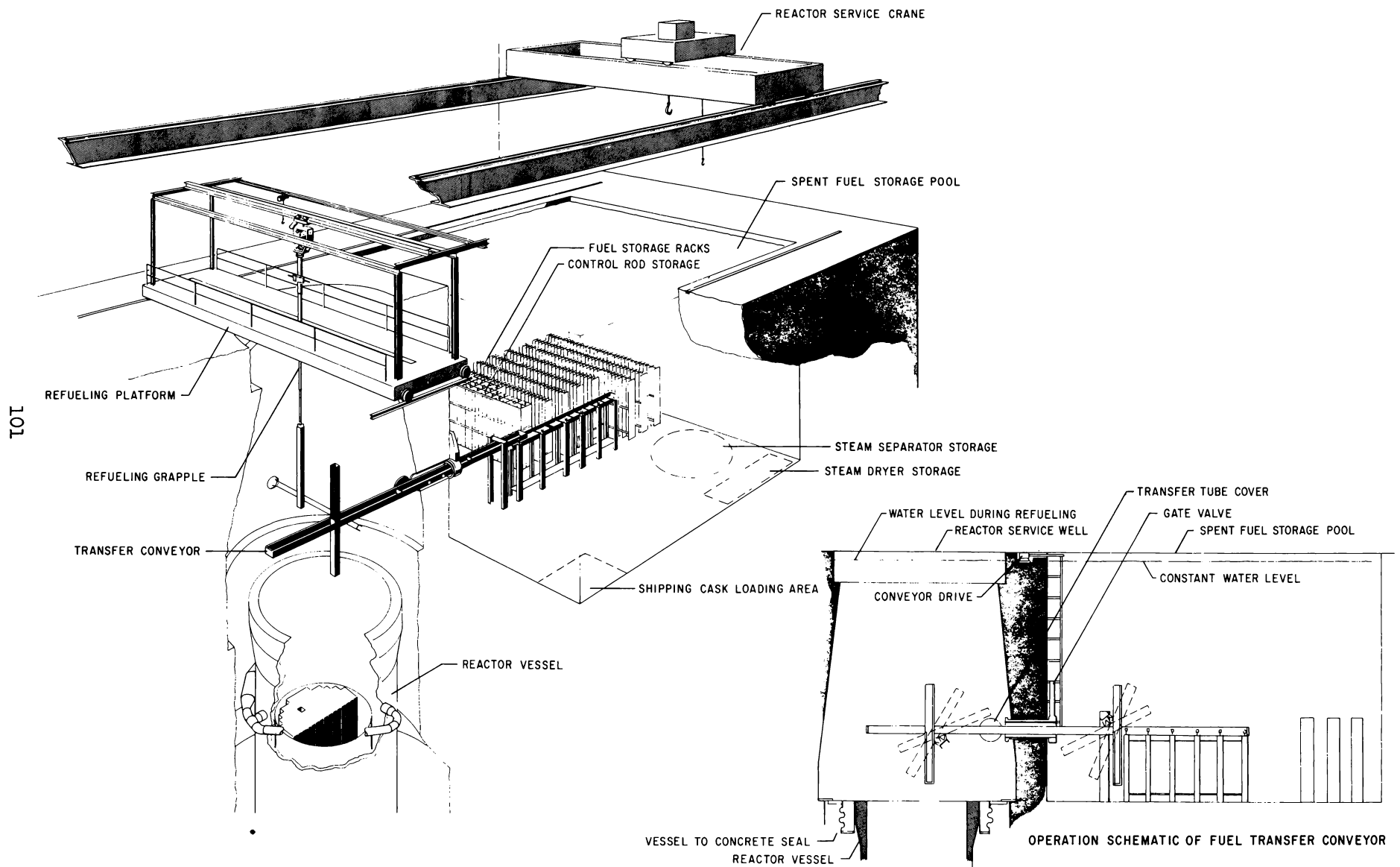
7.2.5 BWR Recirculation Pump Maintenance

A special problem encountered in the design of the proposed plant was the necessity of incorporating the reactor recirculating loops within the boiling water reactor dry well. These pumps must be serviced periodically, and consideration must be given to easy access. This problem has been studied by General Electric, and a suggested solution is offered as outlined below.

A personnel hatch has been provided in the dry well to allow easy access for maintenance personnel to perform routine inspections and minor maintenance during plant outages. (Access is, of course, prohibited during operation.)

If it becomes necessary to make major repairs, a large equipment hatch is provided. Access to this hatch is obtained by removing stacked shielding blocks outside the dry well. The size of the equipment hatch allows the largest single component of the recirculating pump assembly to be removed from the dry well. Once outside the dry well, the equipment can be lifted to grade elevation by the building service crane.

REFUELING SYSTEM



8.0 Safeguard Considerations

8.1 Introduction

The basic safeguard philosophy guiding the design of the plant is summarized in this section. The embodiment of the safeguard philosophy in specific design is reflected in other portions of the report.

8.2 Safeguard Objectives

The safeguard objectives guiding the design of those features important to the safety of the plant are as follows:

- (a) 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 expert recommendations of recognized national and international radiation protection groups.
- (b) Safety against a nuclear accident that might release dangerous amounts of radioactive materials must be preserved even in the event of equipment malfunction, operator errors, or other credible contingencies.
- (c) Confinement of any significant quantity of radioactive materials that might be released from the reactor must be assured in the event a serious credible accident does occur to the plant.

8.3 General Safety Features

Safeguard provisions can be considered in three broad categories: control over receipt of radiation and release of radioactive material during normal operation; accident prevention; and mitigation of effects of credible accidents.

8.3.1 Normal Operation

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

- (a) The design of the gaseous and liquid waste handling systems in such a manner as: to minimize the quantity of these wastes to be routinely released to unrestricted area; to provide for adequate monitoring and measurement of the radioactive content of such materials; to control their release based on measured results.
- (b) Location or shielding of sources of radiation inherent in the system to the extent necessary to minimize personnel exposure during the performance of normal operating tasks in the plant.

8.3.2 Accident Prevention

Features important in preventing serious nuclear accidents include:

- (a) The inherent safety of light water moderator reactors, that is, the negative void and temperature coefficients of reactivity and the strong negative Doppler coefficient of reactivity in the low enriched fuels. These are of particular importance in limiting the extent of such accidents as the start-up accident or refueling accident.
- (b) Two separate and independent mechanisms to assure shutdown of the reactor. The principal mechanism involve control rods capable of fast automatic shutdown of the reactor from any one of several potentially unsafe operating conditions. The other involves a liquid poison system to act as a back-up source of negative reactivity.
- (c) Alternate systems for emergency removal of reactor heat. The primary full power heat removal system is the turbine main condenser. This unit may be used to remove reactor heat under many emergency conditions but in the event the reactor is isolated from the condenser other means are required and provided to remove reactor decay heat. These systems include: an emergency condenser located within the reactor building, and a core spray system design to cool the core sufficiently to prevent fuel meltdown following scram and loss of all other cooling measures.
- (d) A reactor safety system with sensing devices to detect and prevent potentially unsafe operating conditions from becoming too severe. This system includes a sensing device for all reasonably conceivable unsafe operating conditions that might arise from operator errors or equipment malfunction.
- (e) The design of control of the reactor and primary equipment in such a way as to minimize the possibility for operator errors or that the malfunction of equipment could lead to an unsafe operating condition.

8.3.3 Accident Mitigation

Though the likelihood of a serious nuclear accident is extremely remote by virtue of the foregoing design features and the availability of strong procedural control, the protection of the health and safety of the public is further assured by:

- (a) The provision of constant monitoring and automatically controlled isolation equipment on the primary system off-gas discharge line, and

- (b) Housing the reactor and its principal auxiliaries within a vapor tight enclosure.

8.4 Safeguards Evaluations

The fundamental purpose of safeguards evaluations during the conceptual stages of a design study is to show that a plant can be designed to be safe.

8.4.1 Boiling Water Reactor

The design of the boiling water reactor is similar, when viewed from a safety viewpoint, to previous General Electric boiling water reactors. The maximum credible accident is unchanged; this accident is discussed in Section 8.4.3.

8.4.2 Superheat Reactor

The superheat reactor has been designed with consideration of four accidents. These accidents are the hot flooding accident, the pipe rupture accident, the loss of steam flow accident, and the cold start-up accident. As reactor core design and reactor core safety are so closely related, these accidents have been discussed under Section 2.2.5, Reactor Core Design.

8.4.3 Maximum Credible Accident

It should be recognized that because of the relative degree of detailed studies and the state of technology for the superheat concept, that safeguards analysis for the superheat concept must be considered preliminary. It is assumed that the maximum credible accident for the superheater is the same as for the boiling water reactor.

The maximum credible accident involves an instantaneous severance of the largest water or steam line in the primary system while the reactor is operating under conditions of maximum energy content in the coolant.

The pressure suppression containment has been designed for this accident. The liquid coolant of both reactor vessels is assumed to be released. This energy is released to the pressure suppression pool.

In summary, it is believed that this boiling water separate superheat plant can be designed as a safe plant.

9.0 Projected Economic Potential for 1975 Separate Superheat Reactor Plants

9.1 Introduction

The potential of the Separate Nuclear Superheat concept to produce economic power is based on three key features. These are improved technology in the superheat reactor, direct incorporation of advanced boiling water technology and adaptability for very large central station ratings. In consideration of projected economic potential for 1975, advantage of each of these features can be logically incorporated.

9.2 Improved Technology in the Separate Superheat Reactor

Technical improvements in Superheat reactor technology are expected to occur due to normal evolutionary advances as operating experience is obtained. The proposed development program is intended to provide the broad base of nuclear technology as required to construct the 300 MW(e) plant; however, realization of the full potential of the superheat reactor concept to improve the competitive position of nuclear power is dependent upon the degree of success in extending technology in the following areas.

9.2.1 Fuel Costs

Improved fuel cycle costs through reducing the stainless steel to fuel ratio in the active core by thin cladding techniques or by introduction of a low thermal cross section, high temperature cladding material. For long range projections, it is expected that fabrication costs will be reduced to the point, where further reductions in fuel cycle costs are dependent upon increased thermal efficiency and improved neutron economy. In the 1975 projections, no advantage is taken in terms of reduced fabrication costs or utilization of low thermal cross section material. Reduction in fuel costs is shown as a result of improved turbine conditions.

9.2.2 Higher Thermal Efficiency

In the superheater reactor design for 1967 construction, the design limit was fixed at 1250F maximum steady state surface temperature. It is reasonable to expect that with reactor and fuel operating experience it will be possible to raise the average exit temperature to 1050F without a large increase in capital cost of the superheat reactor. This could be done in the same superheater by increasing the ratio of power between the superheater and boiler by accepting either a higher allowable fuel surface temperature or by reducing the ratio of peak to average surface temperature by improved orificing, reducing local power peaking or improving gross power distributions. The expected improvement in energy conversion by increasing the superheater exit temperature to 1050F is about 1.4% or to about 39% gross thermal efficiency. This has the effect of reducing the fuel cost by 3.7% and should also reflect further reductions in the capital cost of the equipment outside of the steam generating equipment.

9.2.3 Increased Volumetric Heat Release

In a steam cooled reactor, the heat transfer properties of the steam are not as good as if water were used as a coolant. On the other hand, there is no heat transfer discontinuity in steam which corresponds to the burn-out heat transfer limit in boiling water reactors. This means that as higher temperature cladding materials become available it will be theoretically possible to operate superheat reactors at higher maximum heat flux than boiling water reactors. The volumetric heat release of the reactor core is proportional to the average heat flux so that in addition to raising maximum allowable fuel surface temperature and maximum heat flux, there is a strong incentive to reduce the ratio of peak to average temperature and power in the superheater. This can be done by flow orificing, local flux flattening and variable moderator or variable enrichment schemes of gross power flattening. Once a surface temperature limit has been established, it is necessary to select the best compromise between increasing reactor exit temperature and operating at increased heat flux to reduce the reactor size. The design of the projected 1975 superheat reactor is based on utilization of higher allowable surface temperatures and improved flattening techniques such that 1050F superheat steam and 50 kw/liter operation is feasible.

9.3 Improvements in Boiling Water Reactor Technology

Since approximately two-thirds of the power for a superheat plant is generated in the boiler, it is apparent that the thermal and economic performance of the boiling water reactor has a dominant effect in the cost of power from a nuclear superheat plant. The projected characteristics of the 1975 plant shown in Table 9.1 are based on the development of a high power density boiling water reactor. The major technical objectives involved in high power density boiling water reactors is to increase the volumetric heat release of the fuel and the coolant.

Increased specific power of the fuel has two aspects. The first is economic in that with reduced fabrication costs, smaller diameter fuel rods may be used. The higher surface to volume ratio of small rods permits operation at higher heat flux. The specific power of the fuel may also be increased by operating at higher UO_2 temperature or by increasing the conductivity of the fuel. The second aspect is burnout heat transfer limits. It is expected that better heat transfer data may permit operation at reduced factor of safety to burnout.

Increased volumetric heat release to the coolant involves a larger enthalpy rise or increased exit quality from the boiling core. Increased exit quality is dependent upon a better understanding of limits in terms of both hydraulic and nuclear stability. For the boiling water reactor projected to 1975, the power density is 75 kw/liter. It is significant that the Consumers Power high power density reactor also has 75 kw/liter reactor power density as a technical goal. The boiling water reactor to deliver 1360 MW(t) for the projected plant is 11 ft. inside diameter. This is the same size vessel as the 619 MW(t) boiling water reactor for the 1967 plant.

TABLE 9.1

COMPARISON OF PLANT CHARACTERISTICS

	<u>1967 Plant</u>	<u>1975 Plant</u>
Net Output (MW(e))	300	750
Cycle Efficiency (gross)%	37.6	39
Thermal Power		
Boiler (MWt)	619.2	1360
Superheater (MWt)	213.8	640
Total	832.0	2000
Turbine Conditions		
Throttle Temperature	900	1050
Throttle Pressure, psia	965	1000
Vessel Diameters ID (ft)		
Boiler	11.0	11.0
Superheater	8.5	10.0
Core Equivalent Diameter		
Boiler	8.5	8.5
Superheater	6.0	8.0
Power Density kw/liter		
Boiler	48.6	75
Superheater	36.6	50
Fuel Cladding		
Boiler	Zr	Zr
Superheater	SS	SS
Fuel Specific Power kw/kg		
Boiler	20	31
Superheater	20	27.4

Although the thermal rating increased by over 100%, no increase in vessel size is required. This projection is based on extrapolations of current data on steam separators now under development. It is noted that increasing exit quality from the boiling reactor reduces the flow requirements on the steam separators.

9.4 Adaptability for Very Large Central Station Ratings

The historical trend in the electrical utility industry has been to construct larger and larger central station units. At the present time, power plants in the 400-600 MW range are under construction, and 750 MW(e) units will be commonplace in 1975. The utility industry has doubled in size every 8-10 years with the size of individual generating units following this pattern of growth. In order to be competitive with fossil fueled plants in the period of 1970-1980, it would seem fairly certain that saturated turbine conditions must be replaced with superheated turbine conditions. The projected 750 MW unit will employ reactor vessels, buildings, and auxiliaries which are nearly the same as the equipment specified in the 1960 300 MW unit. The capital cost of the nuclear steam supply system in terms of \$/kw would decrease by 30-50%. This capital cost reduction of large units is one of the most promising features of the separate superheat reactor concept.

10.0 Research and Development

10.1 Introduction

This section will scope a research and development program required to achieve the stated goal of operation of a large superheat plant by June, 1967. The establishment of an end date requires that key events and the level of effort for the R&D program be fitted to a somewhat demanding schedule. Assuming a three-year construction schedule, and further assuming that the program is not authorized until early in fiscal year 1962, approximately two and one-half years are available for R&D prior to start of construction.

Research and Development is recommended in six major areas:

- 1) Fuel Development
- 2) Core Development, Physics and Safeguards Analysis
- 3) Heat Transfer and Fluid Flow
- 4) Alternate Materials, Corrosion and Coolant Chemistry
- 5) Reactor Control
- 6) Mechanical Development

The recommended program, together with existing R&D programs, must provide a sound technical foundation upon which to base the final design. Major requirements which the integrated programs must supply are:

1. The necessary test equipment and facilities required to develop and proof-test the conceptual design. Due to the long lead time required to design, fabricate, and construct complex facilities, early action must be initiated to insure the availability of the required facilities for the proof testing of final equipment and component designs.
2. The major design parameters and sufficient information to prepare a Preliminary Hazards Summary Report. It is assumed that major design details, based upon the operating experience gained by the operation of test and prototype facilities and acceptable analytical techniques, will have to be presented prior to receipt of a construction permit.
3. Information on which to base the selection of reference and alternate materials and components. The lead time required to manufacture large hardware items necessitates early "freezing" of design specifications for the reactor vessel, turbine, large pumps, etc. In the special nuclear areas such as the fuel elements, control rods, and control instrumentation, a reference design and material may be selected and developed and, in some areas, development of other promising candidate materials or components will be required as a back-up effort.

The recommended program advocates specific rather than broad development. It provides an optimum combination of development, testing, and plant construction such that minimum cost will be expended during the period of pre-economic nuclear power. It provides for reasonable financial and technical risks and permits the full utilization of normal evolutionary advances during sequence of development test, prototype construction, and large plant construction. The program provides the following sequence of events:

1. Establishment of a Technical Base for Nuclear Superheat

This phase of the work was initiated on June 30, 1959, on Contract AT(04-3)-189, Project Agreement No. 13 between the Atomic Energy Commission and the General Electric Company.

2. Construction of a Small 10 to 15 MW Thermal Separate Superheat Fuel Test Facility at the Vallecitos Laboratory

It is anticipated that this facility will be designed and constructed by the General Electric Company in conjunction with a utility group. The cost of the development, operation, and fuel is included as a part of this program. Although the capital costs of providing this facility are not part of the cost estimate for research and development, the facility would be available for testing and determining the performance limits of economic superheat fuel types under this program.

3. Construction of a 300 MWe Boiling Water Separate Superheat Plant

This plant, when constructed, would approach economic nuclear power. The technical information obtained from the earlier phases of the superheat development program and from the boiling water reactor program would result in incorporation of the latest and most advanced features of both reactor types at the time of construction.

The development work associated with the construction of this plant would be limited to fuel, core structure, control rod drives and refueling systems as required to demonstrate reliable, long life performance.

Although the primary motivation for the superheat R&D program is economic, it is anticipated that several other benefits to the overall nuclear program will be realized:

- 1) The realization of nuclear superheat will indicate to potential users of nuclear steam a coming age of nuclear power. Nuclear power will have developed to the point where it can match the requirements of the most advanced steam utilization equipment.
- 2) The fuel development will provide basic technology directly applicable to other reactor concepts and designs. The rigorous requirements for superheat fuel should allow for the adoption of materials, fabrication techniques and designs to other systems with the potential of improved performance and longer life.

The total cost of this program is estimated to be \$20,433,000. The details of the cost estimate are provided in Section 10.6.

The schedule as discussed in Section 10.4 is based on the availability of the required test equipment and facilities as well as the direct financial support.

10.2 Nuclear Superheat Development Problems

10.2.1 Major Problems of Technical Feasibility

The two major problems of technical feasibility are the verification of acceptable radiation levels in a direct-cycle nuclear superheated steam system, and the availability of fuel performance data as required to establish reliability and statistical performance limits on superheat fuels which have economic potential. These questions are related since it is expected that the fuel cladding corrosion, erosion, and transport of fission products will be related to the factor of safety used in the fuel design. The premium is on the determination of the optimum compromise between the required fuel performance and low fuel cost.

10.2.2 Nuclear Superheat Fuel Design

The most obvious conflict of superheat fuel requirements is related to the compromise between the physics optimum and the mechanical and heat transfer practicality. This leads to the requirement of providing the absolute minimum amount of stainless steel in the active core region which is consistent with the required reliability of the superheat fuel. Several practical considerations impose the major limitation on the achievement of the physics optimum. This is the requirement for cladding the fuel which results from the necessity of preventing the coolant from coming in contact with the chemically active uranium and from keeping the fission products and the fissionable material within the controlled confines of the reactor vessel. Utilization of an oxide of uranium minimizes the extent of further chemical reactions; however, utilization of the fuel impose the requirement that the cladding act as a pressure vessel for the fission product gases. In addition, the cladding must act as a pressure vessel in order to prevent collapse of the cladding when subjected to the primary coolant system pressure inside the reactor. For temperatures of interest to nuclear superheat, the corrosion rate and strength of the known low cross-section materials, such as aluminum and zirconium, are not satisfactory for use in a superheat reactor. As a result, high nickel alloys and stainless steels must be used. Since these materials are significantly more of a thermal neutron poison, the neutron economy of a superheat reactor is lower than that of a zircinium-clad boiling water reactor. Although it is desirable to select standard materials fabricated with standard production methods, the incentive for designing in-core components to the most stringent specifications, to minimize the amount of high

cross-section material, may well be worth the extra cost of fabrication and quality control. In addition to the strength considerations for the cladding, an adequate margin for corrosion and erosion must be provided. Provision must be made to insure that thermal stresses and thermal cycling do not cause fatigue failure of the cladding structure. In order to provide for adequate heat transfer characteristics in the steam cooled system, small hydraulic diameters must be utilized. This imposes an additional requirement of fuel element dimensional stability during all conditions of reactor operation.

The achievement of high turbine inlet temperature and high thermal efficiencies provides a means for the reduction of fuel cost, even in a system of relatively poorer neutron economy. As a result, there is significant incentive for gradual increase of the outlet temperature from the superheater. Two methods of increasing the average outlet temperature for the superheater are available.

The most obvious method would be to utilize higher surface temperatures for the fuel. However, surface temperatures of the cladding above 1250°F and 1300°F are questionable due to the rapid deterioration of the physical properties of stainless steels above these temperatures. The second means available for increasing the superheat outlet temperature is to reduce the peak-to-average power and temperature ratios in the reactor so that all the superheat fuel is operating at or near its maximum allowable level. The order of magnitude of gain by this approach can be appreciated by considering that a reduction of peak-to-average from 3.5 to 2.5 would result in a 40% increase in reactor power from the same vessel with the same surface temperature limit.

In order to design fuel intelligently for superheat application, it is necessary to test superheat fuel to failure in order to make a rational determination of the factor of safety of the particular fuel element when operating under its design conditions. This is further complicated by the fact that there are so many variables to be considered in fuel performance. As a result, individual fuel tests may not adequately determine fuel element failure trends. This imposes the requirement of obtaining statistical fuel performance in order to establish fuel performance reliability on a realistic basis.

10.2.3 Coolant Chemistry and Radiation Levels

Conventional superheat power plant technology has produced a family of components and materials which have been proven by use over a number of years. In the majority of cases, these materials and components are readily adaptable to a nuclear superheat plant. However, each material or component must be examined for compatibility with three additional requirements:

- 1) Ability to withstand radiation-induced aging
- 2) Ease of maintenance under radiation conditions
- 3) Provision for decontamination without deteriorating effect

While many problems in this area have been solved in the design and operation of boiling water reactors, some effects may be accentuated by the superheat environment. In a boiling water reactor system, the boiling mechanism tends to separate impurities and corrosion products out of the steam. This decontamination factor is on the order of 1000 to 10,000 in a boiling water reactor system. Since the superheat coolant is single-phased, there is no similar decontamination mechanism in play. For this reason, it may be expected that crud deposition and radioactive material carried over may be more significant in a steam-cooled reactor than in a boiling water reactor. In addition, two other factors may lead to more difficult coolant chemistry problems. There is the possibility of higher corrosion rate and more crud transfer than in a boiling water reactor system. The higher corrosion rate is expected due to the higher cladding surface temperature, superheated steam environment, and the presence of radiolytically decomposed water gases. Increased transfer and deposition of corrosion products, steam system impurities, and other radioactive materials may result from erosion due to high velocity steam and due to thermal shock of fuel element cladding as a result of system temperature transients. Elimination or reduction in the amount of high activity elements such as cobalt may be an aid in minimizing the problem. In addition, it is expected that in the event there is a fuel element rupture, water-soluble fission gas products which would normally be retained in a boiling water reactor system would be carried over and deposited in the primary coolant system of the superheated steam power plant. This would result in a different distribution of fission products in a superheated steam system than has been observed in boiling water reactor systems. It is expected that the activity levels in the primary coolant system and in the turbine would not be any higher at full power than in a boiling water reactor system. This is due to the significantly higher, but short life gamma activity from nitrogen 16 in the primary coolant system. In the event of fuel rupture, the deposition of fission products in the primary coolant system and in the turbine may lead to more difficult decontamination procedures and higher activity levels during periods of maintenance after the plant is shut down. This may require the addition of auxiliary systems which are not required on boiling water reactor plants.

10.2.4 Physics Considerations

In general, a water-moderated nuclear superheater is quite similar in nuclear characteristics to a water-moderated and cooled boiling water or pressurized water reactor. It is the small differences created by the steam flow paths in the reactor core which modifies the physics analysis and requires development beyond that previously done for boiling water reactor cores. The steam void space introduces a more basic physics problem which is associated with the changes in reactivity due to flooding of the steam channels. It is very difficult to design a core in which a reactivity effect of flooding the steam channels is negative and at the same time the moderator void coefficient is negative. In a typical design case, the flooding coefficient is negative when cold and positive when the moderator is hot. This leads to a positive void coefficient cold and a negative coefficient hot. The possible hazard associated with boiling the moderator while cold or flooding the steam channels while hot must be carefully considered in a design of the superheat reactor.

10.2.5 Nuclear Superheat Safeguards Considerations

A detailed preliminary safeguards analysis for the nuclear superheat design appears in Section 2.2.5.5. However, the research and development program is heavily oriented to providing data, information and experience which will satisfy the questions which arise in the course of the hazards evaluation. Therefore, a brief discussion of need for detailed consideration of safety problems as part of the R&D program is provided.

The following is intended as a brief summary of difference in safety and accident problems between a boiling water reactor and a steam cooled nuclear superheat reactor. All of the major differences are associated with the expected differences between a water cooled and gas cooled reactor. For the superheat reactor, since it is desirable from a cycle efficiency point of view to minimize the loss of heat from superheated steam to the moderator there is no prompt temperature coefficient, with the exception of Doppler coefficient, between power level in the reactor and reactivity. This constitutes a major change in control philosophy since in a superheat reactor there will probably not be a built-in power limiting characteristic corresponding to steam voids in the boiling water reactor. In the superheat reactor, it will probably be necessary to adjust reactivity as a function of superheat exit temperature. This imposes a requirement for continuous duty control drive mechanisms.

The superheat reactor has a large number of the safety problems which characterize gas cooled reactors. These are associated with the necessity of providing both heat transfer mechanisms and a heat transfer sink for the reactor fuel under all conditions. Since the volumetric heat capacity of steam is not large enough to utilize natural convection cooling, it is necessary to provide a reliable means of maintaining a heat transfer mechanism from the fuel to the heat sink. This problem is simplified in the boiling water reactor since the moderator water provides the dual function of heat transfer medium and heat transfer sink. In the case of steam cooled superheat reactors, this function is provided by steam from the boiling water reactor. For emergency conditions, direct thermal radiation from the fuel element to the process tube separating the steam flow passage from the moderator may be utilized. Preliminary studies have indicated that even with the direct thermal radiation heat transfer mechanism, a relatively large flow of steam is required for several seconds in order to remove the stored heat and decay heat from the fuel element.

The economic requirements for good neutron economy are in conflict with the requirements for minimum reactivity change in the event that the passages provided for steam flow become filled with moderator water. Studies have been made which indicate that large changes in reactivity are possible under certain conditions unless the superheat reactor design is adjusted to minimize these effects. The change in reactivity on flooding can be made small by reducing the steam flow passages, utilization of soluble poisons in the moderator, or reducing excess reactivity.

An additional problem of a steam cooled annular fuel element is the possibility of melting the inside fuel cladding which would result in a reactivity increase. The magnitude of this increase may be reduced by utilizing thin clad fuel elements and by arranging the reactor design such that the reactor would be scrammed immediately on anticipation of any accident which might cause melting of the inside diameter fuel cladding.

The radiation levels expected from a single phase gas cooled reactor in a direct cycle system are different than one would expect from a boiling water reactor. This difference is due to the fact that in a boiling water reactor, due to the phase change, or evaporation effect, there is a decontamination factor which may be on the order of 10^3 to 10^4 . In the case of a single phase steam cooled reactor there is no similar decontamination mechanism in effect. For operation with sound fuel elements, the activity level may be higher due to higher steam temperatures, presence of radiolytically decomposed hydrogen and oxygen, possibility of thermal shock of fuel surfaces which would result in high activity "crud" brusts and possible increase in erosion of fuel element surfaces due to high velocity steam. For operation with a defective fuel element, the activity levels may be higher due to the fact that there is no decontamination effect in the steam cooled reactor system. This may result in more particulate matter carried over into the turbine and also the presence of certain fission product materials, such as Iodine which being soluble in water are not normally present in the exit steam from a boiling water reactor operating with defective fuel. The limited experimental data obtained from the SADE loop has verified the presence of Iodine in the SADE discharge steam piping. Analytical predictions indicate that the steady state turbine activity due to carryover of Iodine-131 from a defective superheat fuel element will not be significantly larger due to the high background activity of N-16. It is expected that some auxiliary system, such as silver mesh, may be required to scrub Iodine out of the stack. In addition, it is expected that the background activity after shutdown may be higher in a superheat turbine. It is predicted, however, that because of the relatively short half life of the Iodine-131 that normal detergent cleaning will remove the major portions of this contamination.

10.2.6. Heat Transfer

In any gas-cooled reactor system, the coolant by virtue of its low volumetric heat capacity, imposes a serious design problem associated with providing a continuous and reliable means of providing coolant flow at all times. This problem is significantly more difficult than in a liquid metal or water-cooled reactor system, where natural circulation cooling may be utilized. In a gas-cooled reactor, some provision must be made to provide a positive heat sink, either by auxiliary or back-up coolant systems, or by utilizing a secondary heat transfer mechanism. In a light-water-moderated system, the moderator constitutes a very large heat sink. In the General Electric Company superheat reactors, the utilization of the direct thermal radiation mechanism for loss of coolant flow either during scram or decay heat removal operation, provides a reliable, practical solution to this problem.

10.2.7 Control of Nuclear Superheaters

While instrumentation exists for both superheated steam conventional power plants and boiling water reactors, the requirements for a nuclear superheater pose a number of development problems. Special detectors are required for steam temperature, fuel temperature location of ruptured fuel elements, incore flux patterns, incore steam flow, etc. Based on present information, there are no completely acceptable neutron detectors which will operate reliably for extended periods in a steam environment. The higher temperatures result in special materials problems. Since the safety of the reactor is to a large degree dependent on the accuracy and reliability of the instrumentation, a thorough testing program must support the selection of new or modified sensors and instruments.

A separate nuclear superheater, connected in series with a boiling water reactor, reduces the control problem. As in any boiling water reactor system, the system pressure must be maintained constant by varying reactor power to correspond to load. Superheat reactor power will be adjusted to provide constant superheat outlet temperature. Since there is no nuclear coupling between the boiling section and the superheat section, reactivity can be controlled by control rods. This will require a continuous duty type control mechanism since there is no, or little, built-in load following characteristics.

The problem of temperature sensing devices, and particularly the time delay associated with these mechanisms, may present design difficulties. With this type of system, it will be required that the superheat reactor never become critical before the boiling water reactor has warmed up and is generating steam. In a similar manner, scram of the boiling water reactor would also scram the superheat reactor. With this condition, a limited supply of steam would be available to the superheater because of the stored energy in the boiler. For slow load changes where the boiling water reactor pressure is maintained constant, the power level in steam flow from the boiling water reactor would match load demand. For fast transients where load is dumped and an attenuator for by-passing the superheated steam to the main condenser may be required. The ability to pick up load rapidly may be somewhat slower in the boiler-superheat plant because of the increased system time constants; however, it is expected that this will not constitute any unusual operational limitations.

10.3 Detailed Research and Development Program

At the present time there are no superheat reactor plants in operation. In addition, there is little irradiation data on nuclear superheat fuels. However, the satisfactory operation of the SADE loop in the VBWR has indicated that there are no insurmountable technical problems which would prevent the utilization of superheated steam as a reactor coolant. The major technical risk, therefore, is the uncertainty associated with the ability of the superheat concept to produce power at a lower cost than an advanced boiling water or other reactor system.

The light water-moderated steam cooled superheat reactor provides the promise for the earliest exploitation of nuclear superheat. At the present time there is not enough information to provide the basis for a clear selection of integral vs. separate superheat concept. A development program including a separate superheat reactor is not incompatible with integral superheaters; on the contrary, development of the separate superheat reactor will allow for pinpointing problems common to all superheat reactors which will, in turn, allow for early solution and orderly advancement in the state of the art.

A research and development program to result in the operation of a 300 MWe superheat plant in June, 1967, contains four elements as follows:

- 1) Base superheat development program
- 2) Construction and operation of a flexible fuel test facility
- 3) Specific research and development, design, construction, start-up, and operation of a 300 MWe superheat plant
- 4) Confirmed operation of the fuel test facility, prototype superheat reactor and large superheat reactor to develop the additional technology required to exploit the concept.

10.3.1 Base Development Program

Contract AT(04-3)-189, Project Agreement #13

This phase of the work was initiated on June 30, 1959, between the Atomic Energy Commission and the General Electric Company. The objective of the nuclear superheat project is to establish a valid experimental base leading to realistic nuclear superheat reactor design criteria which will recognize the potential for evolutionary improvements, and ultimately realize the cost advantage of higher thermal efficiencies. This program is intended to establish on as realistic a basis as possible the order of magnitude of difficulty of development problems and the potential of nuclear superheat to improve the competitive position of nuclear power. The nuclear superheat project is a coordinated, flexible program to investigate in depth the major problems of technical feasibility for the economic application of nuclear superheat for power generation.

The base development program is considered to apply to no one superheat concept. Therefore, it is considered inappropriate to charge the cost of this work to the development program for the 300 MWe plant.

Major tasks and significant progress to date are as follows:

10.3.1.1 Task A - Conceptual Design and Program Evaluation

The purpose of Task A is to provide the engineering design and analysis, nuclear physics, and reactor systems evaluations that are required to establish economic potential, development program test conditions, and nuclear superheat reactor design criteria. This work is needed in order to establish realistic development test conditions, to recognize design areas most significant to the reduction of power generation costs, to emphasize these areas in development testing, and to establish realistic reactor design criteria based on development results.

The set of design parameters for the 300 MWe Separate Superheater plant has been established as a result of work on this task.

10.3.1.2 Task B - Fuel Technology

The purpose of the fuel technology program is to design, fabricate, irradiate, and evaluate various superheat fuel concepts, including various fabrication techniques, various geometries, and variable fuel cladding thickness. The need for this activity is to provide realistic design information on superheat fuel in order that an intelligent selection can be made of the minimum factor of safety and maximum fuel operating conditions which are consistent with the high degree of reliability required from power-producing nuclear systems. The determination of operating limits and fuel performance is the most important single aspect of developing nuclear superheat technology. The major engineering development tool for the fuel technology program is the irradiation of superheat fuel elements in the SADE loop in the VBWR reactor.

An aggressive development program is well under way and features out-of-reactor heat transfer, dynamic erosion and performance of warped fuel element tests, as well as in-reactor irradiation of capsules and prototype superheat fuel elements in the GETR and SADE loops in VBWR.

Work to date has involved UO_2 pellets and the work must be classified as "screening" and inconclusive. Although no major difficulties have been noted and several promising designs have been evolved, continued emphasis on this phase of the program is required.

10.3.1.3 Task C - Materials Development

The purpose of the materials development activity is to provide materials property evaluations on stainless steel fuel cladding. Work to date includes the determination of yield strength and ultimate strength on several cladding specimens taken from fuel elements irradiated in the SADE loop. In addition, analytical studies and experimental evaluations will be made to predict the cyclic strain and fatigue characteristics of cladding materials operating in the plastic region. The need for this activity is based on the apparent necessity of minimizing the amount of stainless steel in the superheat core in order to improve the neutron economy and achieve low power cost.

10.3.1.4 Task D - Experimental Physics

The purpose of the experimental physics program is to determine the limits on questions of basic feasibility for the annular superheat fuel geometry and a light water matrix. This work is needed in order to establish the validity of current nuclear physics calculational methods for annular fuel geometries. The probability of reactivity changes due to flooding of steam coolant passages and reactivity changes associated with temperature effects in the moderator are very significant to both reactor safety and also to achievement of high volumetric heat release from the superheat reactor.

The critical experiment facility at Vallecitos Atomic Laboratory has been designed, fuel fabricated and experimental measurements started during October, 1960.

10.3.1.5 Task E - Corrosion and Coolant Chemistry

The purpose of this activity is to determine the expected distribution of radioactivity in the main steam system of an operating direct cycle, nuclear superheat plant. The work involves theoretical analysis and experimental evaluations, both in-pile and out-of-pile, to determine the corrosion rate, radioactive deposition, and expected radiation levels for both sound and purposely defected superheat fuel. The need for this activity is associated with providing fuel design information as required to anticipate corrosion and erosion difficulties, and also to determine any system requirements that result from the use of superheated steam in a direct cycle system.

A dynamic corrosion test loop has been erected and is in operation. Work to date has been in support of the in-reactor operation of the SADE loop.

10.3.1.6 Task F - Heat Transfer

The purpose of this activity is to provide the engineering analysis and experimental evaluation that is required to determine heat transfer limitations associated with high steam quality, and to provide engineering design data for use on the design of reactors using superheated steam as a coolant. The need for this activity is based on the fact that even though steam is a conventional heat transfer medium, there is not adequate experimental data in several areas of interest to superheat application. This program will provide experimental data on heat transfer in regions of high steam quality, and heat transfer in regions of high mass flow rates.

An out-of-reactor "burnout" loop has been operated and produced data useful to the establishment of conceptual design parameters. Work is proceeding on the instrumentation of in-reactor fuel bundles to provide definitive heat transfer performance data.

10.3.1.7 Task G - Mechanical Development

The purpose of this task is to provide the engineering analysis and experimental verification of the predicted performance of several mechanical components of the nuclear superheat reactor such as steam-water separators, seals, design of large tube sheets and nozzles, and materials compatibility. The need for this activity is based on establishing the performance characteristics, life and reliability characteristics of mechanical components, which are essential to establishing low cost and reliable performance from nuclear superheat reactors.

Work to date has been centered on the selection and testing of candidate steam separator designs.

10.3.1.8 Task H - Superheat Advance Demonstration Experiment

The purpose of the SADE loop is to provide the basic engineering development tool for fuel element performance evaluations, corrosion, and coolant chemistry evaluations, and the determination of operational characteristics of superheat systems as required to perform the previously listed tasks. The need for this facility is based on the necessity of obtaining performance data from a single, well-instrumented superheat fuel element, operated under identical conditions to that expected in a superheat steam system. The results from this experiment should require a minimum of extrapolation in terms of establishing realistic reactor design criteria. Obvious limitations of fuel testing in the SADE loop are the relatively low neutron flux, the limitation on the number of fuel elements that may be irradiated, and a conflict between testing a few fuel elements to high burnup, or a large number of fuel elements to a relatively low burnup.

Irradiation of prototype fuel assemblies has been carried out in the SADE loop. The loop has been modified to provide for: a) better control of irradiations, and b) additional performance data. Irradiation of promising fuel designs will continue within the limitations of the loop.

10.3.1.9 Additional Considerations

A proposal has been made to the Atomic Energy Commission to expand the present single fuel element SADE loop to a 9-element expanded SADE loop. The AEC has given technical approval to this proposal and design activity has started. It is expected that the expanded SADE facility will be in operation in the spring of 1961.

A continual review of other superheat programs insures proper direction of the basic superheat program. This integration is provided by the AEC through their semi-annual Nuclear Superheat Technical Meetings.

10.3.2 VBWR Hook-On Fuel Test Facility

The basic research and development program, now in progress, contains all of the elements required for the research and development program for a 300 MWe plant. However, in some areas augmentation and acceleration of the effort is required.

The present research and development program can best be augmented by providing a facility to perform the required development of superheat fuel. The facility will be expected to answer or confirm major questions of:

- Fuel performance
- Fuel corrosion, erosion and radioactivity transport
- Control and load matching of boiler and superheater
- Simplification of reactor and plant design
- Criteria for reactor safeguards requirements

The General Electric Company is designing and constructing a separate superheat facility to be operated in conjunction with the VBWR. The facility which will be devoted exclusively to the development of superheat reactor technology and is not encumbered with a requirement for electric power generation will (1) provide for in-pile testing of a large number of superheat fuels of various designs, and (2) will provide for the development of information important to basic nuclear superheat reactor technology.

In this country and abroad, the value of similar pilot reactor facilities as a means for obtaining basic reactor technology has been demonstrated and the results obtained have accelerated the pace of technological advances.

The General Electric Company has indicated a willingness to make this facility available, on a priority basis, for the Atomic Energy Commission Superheat program. The integrated Research and Development program described in this report postulates the essentially full-time application of this facility to the program requirements.

10.3.2.1 Functional Description of Reactor

The purpose of the VBWR Separate Superheat Reactor is to provide the functional requirement of a superheat fuel element test facility. The initial loading of the reactor will consist of 288 stainless steel clad annular fuel elements similar to those previously developed and proof tested in the SADE loop. Since the reactor diameter is relatively small, with a high peak-to-average distribution, a relatively wide range of fuel performance in terms of heat flux, surface temperature and exposure can be obtained in a relatively straightforward manner. The reactor facility will be provided with steam flow orifices such that variations in heat transfer conditions, steam velocity, and pressure drop may be obtained. It is expected that the initial loading of fuel will provide a significant number of fuel element samples of various clad thicknesses and manufacturing procedure in each power level location. This will provide a basis of direct comparison of important performance variables. The reactor will provide sufficient flexibility so that, if desired, various fuel element configurations of a more advanced type may be tested providing continued utilization of the facility.

From this point of view, the VBWR Separate Superheater may be considered a critical assembly of in-pile test loops. Valving will be provided to permit continued operation with defective superheat fuel.

10.3.2.2 Major Technical Objectives of VBWR Hook-On

The three major technical objectives of the VBWR Hook-on facility are as follows:

- 1) To provide a definitive answer to questions concerning radioactivity levels of a direct cycle superheat steam-cooled reactor system for both sound and defective superheat fuel. It is expected that these evaluations will provide a basis for establishing any auxiliary system requirements for the full size plant.

- 2) To provide an experimental basis for evaluating the performance limits of superheat fuel elements by testing fuel to failure at a relatively wide range of test conditions, such as heat flux, cladding surface temperature, fuel irradiation exposure and superheat coolant conditions of moisture, temperature and velocity.
- 3) To provide an experimental basis for establishing realistic superheat fuel design criteria as required to optimize superheat reactor performance for minimum fuel costs and technical and economic risk.

10.3.2.3 Recommended VBWR Hook-on Phase of the Research and Development Program

The General Electric Company has previously recommended a program of advanced fuel fabrication and testing based on the availability of the hook-on facility. This program is applicable and in direct support of the overall program required for a large Superheat plant.

In general, the R&D program is heavily oriented to the delineation of fuel element parameters with emphasis on multiple in-reactor irradiation of defected as well as sound fuel elements.

The major elements of the recommended program are as follows:

Test Program Planning and Evaluation

Objective - To make certain that all phases of fuel development work performed in the VBWR Hook-on has specific direction toward realistic objectives, and that the design of the VBWR Hook-on facility will provide the test environment required in order to obtain experimentally valid data.

Plan of Attack - Establish the development program test requirements, initiate VBWR Hook-on reactor specifications, initiate test instructions, and evaluate and interpret results.

Fuel Design and Fabrication

Objective - To design and fabricate superheat fuel elements for the first core for the Hook-On Superheat Reactor.

Approach to the Problem - Conduct design studies and analytical investigations as required to design and fabricate superheat fuel elements for both first core fuel types and more advanced development fuel types. Analyze and interpret results from fuel irradiation testing in the SADE facility in VBWR and other AEC programs and incorporate the most promising fuel types into more advanced fuel development testing.

Fuel Irradiation

Objective - To provide an experimental basis for determining the limits of performance of superheat fuel elements.

Plan of Attack - Design, fabricate and irradiate a large number of superheat fuel elements to determine the effect of surface temperature, heat flux, clad thickness, velocity limits, erosion and corrosion and irradiation exposure levels.

Defected Fuel Irradiation

Objective - To determine the release rate from purposely defected fuel elements as a function of heat flux and surface temperature, in order to establish auxiliary system requirements and activity levels in a direct cycle superheat steam-cooled reactor system.

Plan of Attack - Install a significant number of purposely defected superheat fuel elements in the VBWR Hook-on to measure activity levels. Perform irradiation of sound fuel elements until failure in order to evaluate the activity levels from an "in service failure" on the superheat fuel elements.

Radioactive Material Laboratory Investigations

Objective - Perform post-irradiation investigations to determine physical changes in fuel elements, metallurgical changes, and cause of failure of superheat fuel elements. Investigate propagation of failure under actual operating conditions.

Plan of Attack - Perform RML investigations on a number of sound, purposely defected and "in service" superheat fuel elements after irradiation in the VBWR Hook-on.

Superheat Instrumentation

Objective - To develop reliable instrumentation for use in a superheat reactor for determination of steam temperature, fuel surface temperature and location of ruptured fuel.

Plan of Attack - Develop instrumentation required for both fuel irradiation program and for subsequent use in a full size superheat reactor by installation and test of required instrumentation in the VBWR Hook-on.

Hook-on Superheat Reactor Operations

Objective - To operate the Hook-on Superheat Reactor as required to perform the experimental evaluations in Superheat Fuel Development and evaluate the performance of a Separate Reactor.

Approach to the Problem - Provide experienced scientific and technical personnel for operation of a new reactor type as required to insure successful completion of the required work on schedule. In addition, provide the highly specialized supporting scientific personnel and facilities to provide for complete accumulation and evaluation of the performance data.

Superheat Evaluation

Objective - To determine the reactor operating characteristics of a separate superheat reactor such as load following, behavior under transient load changes, decay heat removal characteristics and overall operating characteristics when coupled to a boiling water reactor and a turbine heat sink.

Plan of Attack - Measure reactor characteristics under various load and operating conditions.

10.3.2.4 Expected Results VBWR Hook-On Program

Although the VBWR Hook-on phase of the program is scoped to be of a flexible content and subject to changes in direction as definitive information is generated. It is expected that major technological breakthroughs will result. Examples are as follows:

- 1) Definition of operating limits and fuel element failure modes which may be used to design fuel elements for Superheat reactors which have economic potential.
- 2) Definition of system radiation levels and release rates for failed fuel elements.
- 3) Determination of auxiliary system requirements, if any, for a direct cycle superheated steam cooled reactor system.
- 4) Determination of the maximum number of fuel element failures allowable without causing a reactor shutdown.
- 5) Determination of any special accessibility or maintenance problems in the primary coolant system and turbine of a direct cycle steam cooled system.
- 6) Determination of significance of design variables on superheat fuel performance.
- 7) Insight into mode of failure which will permit improved fuel design.
- 8) Experimental results which will allow prediction of superheat fuel life.
- 9) Minimum of extrapolation of fuel test data required due to close simulation of fuel operating requirements by virtue of boiling water reactor steam source.
- 10) Establish precedent for superheat reactor operating characteristics by actual operating experience.
- 11) Establish decay heat removal characteristics, start-up characteristics and shutdown characteristics by actual operating experience.
- 12) Establish precedent for the safety criteria of separate superheat reactors as required to meet AEC licensing requirements.
- 13) Development of instrumentation as required to obtain and interpret fuel element performance under the development program.
- 14) Development of reliable instrumentation for use in a large superheat plant.

10.3.3 Large Superheat Reactor Development

The Separate Nuclear Superheat plant concept for operation in June of 1967 represents a significant step beyond present technology. This is true not only for the Separate Superheat Reactor, but also for the boiling Water Reactor. The development work for the boiling water reactor technology is currently being conducted through several AEC sponsored development programs such as City of Los Angeles, Consumers Public Utilities, etc. It is of particular significance to note that since approximately two-thirds of the system power is generated in the boiling water region, the economic incentive of the system is strongly dependent upon achieving economic performance from the boiler. In addition, from a development point of view, in the Separate Nuclear Superheat concept, the development advances made in the boiling water reactor may be incorporated directly.

The technical advances incorporated into the large nuclear superheat reactor are based on results from the previous development activities in the base program, and fuel test facility. The development work on the large superheat plant is limited to those areas wherein proof testing is required to provide reasonable assurance of achieving predicted performance in terms of economics, reliability and system maintenance.

Associated research and development for the large separate nuclear superheater is recommended in areas as follows:

1. Fuel Development

Basic superheater fuel technology would have been developed as the result of the several activities in previous development programs such as allowable surface temperatures, velocity limits, cladding corrosion margin, burnup limits, and fabrication techniques. A period of pre-operation R & D in fuels technology would be required. Fabrication development of long superheat fuel elements would be performed. Irradiation tests of these fuel types (in shorter lengths) would be performed. It is expected that the fuel test facility at VBWR will be utilized effectively during this period. The major development effort would be associated with flow and vibration characteristics of individual fuel elements, support methods, etc., of the longer fuel elements which are characteristic of large thermal output reactors.

2. Core Development

Core development activity would be required to establish detail design data and obtain performance and fabrication data for fuel assembly units, grid structure and control rod and drive systems. Four prototype fuel assemblies would be constructed in order to perform the tests outlined below:

- a. Control Rod Tests - A complete control rod and drive assembly will be assembled in a control rod test facility. Tests will be made at operating temperature and pressure to assure conformance to specification, obtain data on performance and reliability and to ensure conformance to life requirements.
- b. Fluid Flow and Vibration Tests (Fuel Assemblies) - A representative superheat fuel assembly will be tested in a suitable facility to determine plenum hot channel effects, velocity distributions, pressure drop, orificing characteristics and freedom from flutter or vibration difficulties.
- c. Fluid Flow and Vibration Tests (Core Structure) - A prototype core assembly will be fabricated and tested in a suitable facility to determine overall core plenum effects and hydraulic characteristics.
- d. Fabrication Investigation for Core Structure - A prototype core assembly will be fabricated to provide development information on fabrication of grid elements, core structure and control rod rubbing surfaces. In addition, certain limited experimental stress analysis work will be performed to confirm analytical methods of several component parts such as tube sheets, etc.
- e. Fuel Handling Prototype - A prototype fuel assembly will be assembled in the core mock-up and fuel handling facility. Development work will be performed to establish design detail of special handling tools and equipment, refueling procedure and thus ensure relative ease of handling during refueling.

3. Post Operational R & D Program

In addition to the normal start-up tests for a first of type reactor, a period of performance evaluation and analysis is necessary in order to obtain design data that is not available from standard plant instrumentation. This information would be required not only to provide a better understanding of nuclear, heat transfer and power level and power transient limits, but also is required to provide the proper direction to refinement of plant design that may be necessary for the concept to achieve its full economic potential in second and third plants of the same concept.

The four phases of post operational R & D on the operating plant are as follows:

a. Start-up Tests

These tests include critical measurements, pump performance tests, control rod tests, low power reactivity measurements, etc.

The cost of these tests are included in the engineering cost of the plant for start-up.

b. Fuel Performance Evaluation

Several specially instrumented fuel assemblies for performance evaluation will be included in the first core. Provision will be made for special non-destructive and destructive radioactive material laboratory measurements on these and other standard fuel assemblies.

c. Coolant Chemistry Evaluations

Provision will be made to evaluate the water chemistry, radiation activity levels and activity distribution of the plant in order to establish performance characteristics and design margin of shielding, coolant purification systems etc. These evaluations will contain over a fairly long period of time in order to establish the rate of activity build-up in the plant and also to evaluate the significance of activity bursts due to in-service type fuel failures.

d. Nuclear Physics and Plant Performance Evaluations

After the plant has started, and is producing power, there will be a continuing effort by an evaluation team to provide feed-back to the reactor manufacturer. This information will be used to coordinate refinements in design of following plants of the same concept.

10.4 Research and Development Program

The target date of 1967 for operation of a 300 MW(e) economic super-heat plant requires that each development activity occur in sequence to permit the timely utilization of the development information for the next succeeding steps. In addition, periodic review of the progress to-date must take place and the direction of the program altered accordingly. At the outset it is expected that the research and development activities in several areas, such as the selection of a fuel rod cladding material, will encompass several materials or concepts. A date must be established at which time the candidate material is selected or the design of a component frozen on other processing materials or concepts; however, the primary effort will emphasize the selected material or component. Therefore, the schedules developed for the program must contain dates of decision as well as dates of accomplishment. Table 10.1 presents the schedule for accomplishment of key phases of the development program.

TABLE 10.1

SEPARATE SUPERHEATER DEVELOPMENT PROGRAM

		C A L E N D A R Y E A R S							
		1961	1962	1963	1964	1965	1966	1967	1968
Base Development Program (Ref. Section 10.3.1)	SADE	xxx							
	E-SADE	xxxxxxxxxxxx							
	HSR Planning	xxxxxxxxxx							
Hook-On Superheat Reactor (Ref. Section 10.3.2)	Design	xxxxxxxxxxxxxx							
	Construction	xxxxxxxxxxxxxx							
	Operation		xxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxx						
	R&D Program	xxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxx							
300 MW(e) Separate Superheat Plant (Ref. Section 10.3.3)	Design			xxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxx			xxxxxxxxxxxxxxxxxxxx	xxxx	
	Fuel Fabrication						xxxxxxxxxxxx		
	Construction			xxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxx			xxxxxxxxxxxx	xxxx	
	Safeguards		xxxxxxx*				xxxxxxxxxxxx	**	
	Operation							xxxxxxxxxxxxxxxx	xxxxxxxx
	Pre-operation R&D					xxxxxxxxxxxxxxxxxxxx			
	Post-operation R&D								
Alt. Development Program Prototype Reactor (Ref. Section 10.5)	Design	xxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxx							
	Fuel Fabrication			xxxxxxxxxx					
	Construction		xxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxx						
	Safeguards	xxxxxxx*		xxxxxxxxxx**					
	Operation					xxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxx		xxxxxxxx	xxxxxxxx
	Pre-operation R&D			xxxxxxxxxxxxxxxx					
	Post-operation R&D								
* Preliminary Report									
** Final Report									

10.5 Alternate Development Approach - Prototype Superheat Reactor

10.5.1 Introduction

The 1967 Operating Date for a large 300 MW(e) Separate Superheat Plant precluded the construction and operation of a prototype separate superheat plant in time for the fuel performance results to be utilized effectively in the design of the large plant. It is clear that the risk in achieving the predicted economic performance of the large plant will be increased to some extent if prototype reactor experience is not available. This is true because unless there is an adequate technical basis for engineering decision, the only recourse is to provide an increased design margin to provide for added contingency. This is particularly true for fuels development where years of fuel operation are necessary to provide a valid statistical basis of fuel performance. Although a realistic schedule does not permit effective utilization of a prototype separate superheat reactor for the design of a Separate Superheat Plant for 1967 operation, a prototype reactor should be considered in the overall superheat development program because of its importance in minimizing technical and financial risk and of extending technology of second round plants of the same concept.

10.5.2 Technical Justification

The VBWR "hook on", supplemented by the SADE loop, will provide the necessary facilities to perform the required development of superheat fuel types. Fuel operating experience and reliability is the key to power cost reductions. This is true from the point of view of industry acceptance of a large plant, and also because the desired cost reduction with size is not valid for high risk plants. The large superheat plant must be beyond the development stage if cost reductions are to be realized through increased size. Operating experience from a prototype test reactor would provide design information on system activity, steam turbine accessibility, and auxiliary system requirements in a direct cycle system using a single phase gas coolant. Operation of a prototype superheat reactor would provide operating experience which would be useful in establishing operating procedures for the 300 MW(e) Separate Superheat Plant and would be of value in further economic refinement of the plant concept. For a large superheat reactor, the high reliability requirements dictate development work for mechanical components which are unique to superheat reactors as compared to boiling water reactors. A prototype reactor would provide invaluable experience on internal steam manifolding, high temperature instrumentation, refueling systems and other auxiliary systems. The fuel performance

and design information from a small separate superheater are meaningful to application for other large superheat reactors. Although the fuel performance requirements are approximately the same for either the separate or integral reactor, the extrapolation of small to large reactor cores in the separate concept is considerably more straightforward because of the constant void in the entire core of the separate superheater.

10.5.3 General Description of Prototype Superheat Reactor

The output rating of the Boiling Water-Separate Superheat Plant should be about 75 MW(e). This size is based on providing the minimum size plant, in order to minimize financial risk, that will still provide statistical fuel performance data and meaningful extrapolation to superheat plants in the economic size range.

The technical requirements for this plant are:

1. Provide statistical performance data on the installed superheat fuel as required to establish fuel integrity at design surface temperature, and burnup, when subjected to erosive and corrosive action of superheated steam coolant.
2. Provide power plant performance data as required to establish plant reliability, operability, normal maintenance, decontamination effectiveness, refueling, and any additional auxiliary system requirements.
3. Provide reactor performance data on fuel, load following characteristics and plant safety as required to establish a safeguards precedent for the separate superheater plant concept.
4. Provide a useful power generating station for continued use after completion of the post operational research and development program.

The above requirements are satisfied in the 75 MW(e) power plant. The significant areas of extrapolation are fuel length and steam flow passages, reactor vessel and coolant nozzle size, and reactor internal core arrangement and manifolding.

The advantages of the 75 MW(e) size as compared to a larger power plant are as follows:

1. In small plant sizes, more advanced or developmental features may be incorporated because of the relatively lower financial risk. Intelligent selection of the minimum safety factor for superheat fuel which is consistent with the high degree of reliability for nuclear plants is the key to the economic incentive of nuclear superheat.

2. Ultra-conservative safeguards features may be incorporated at relatively low cost in order to establish realistic safeguards criteria for larger superheat plants.
3. Due to the high development content in first-of-its-type plants, the usual reduction in cost per installed KW is not applicable for larger plants. This results in a major capital risk in large developmental plants.
4. Power plant down time due to development difficulties or required changes, although higher in terms of operating and capital cost per installed KW, will amount to a lower total cost.

The advantages of the 75 MW(e) size as compared to a small power plant are as follows:

1. The prototype superheat reactor plant must permit meaningful extrapolation of fuel performance, power plant performance, and safeguards performance to large, economic nuclear superheat plants. The 75 MW(e) plant as contrasted to a small power plant will provide sufficiently valid data in order to permit this extrapolation.
2. Although it is recognized that extrapolation is a matter of degree requiring considerable judgment on the part of the designer, the reactor must be large enough to utilize fuel elements representative, except in length, of the full size reactor. In addition, the core must be large enough to reflect power distribution, void content, neutron leakage and other nuclear characteristics close enough to the full size reactor to permit verification of engineering physics calculation methods. The approximate 60 MW(t) superheat reactor for the 75 MW(e) power plant is considered to be the minimum size from this point of view.
3. Since one of the technical requirements for this plant is that it provide a useful power generating station for continued use after completion of the post operational research and development program, the selection of the 75 MW(e) output provides a good probability of meeting this objective.

10.5.4 Development Program Associated with the Prototype Superheat Reactor

The development program which is essential to the successful and meaningful execution of a nuclear superheat demonstration is presented in detail in four major categories. These are fuel development, nuclear development, plant development and operational planning. The fuel development work is intended to establish performance limits, reliability

burnup and fabrication costs of fuel operating in the superheated steam environment. In addition, development work on improved fabrication techniques will be done which will result in reduced fabrication costs. The nuclear development work will consist of investigations of design performance and limitations in the areas of reactor control, flood safety, decay heat performance, transient performance, heat transfer and fluid flow and overall system performance. Inherent in the suggested program is the need for obtaining statistical fuel performance data and operational tests and analyses of the type that can be satisfactorily performed only in an operating reactor.

Fuel Development Program

1. Pre-Operational Phase

- a. Irradiate prototype fuel bundles in SADE and/or VBWR Hook-on refueling characteristics, dimensional stability, and verify nuclear, heat transfer and hydrodynamic predictions.
- b. Examine irradiated fuel elements in RML to determine fission gas buildup, geometric stability of element, cladding corrosion and erosion.
- c. Investigate means of reducing fuel fabrication cost by improving fabrication techniques.

2. Post-Operational Phase

- a. Irradiate experimental fuel under predicted conditions, evaluate performance and extrapolate results to higher performance fuel and core arrangements.
- b. Fabricate, install and irradiate more advanced fuel concepts.
- c. Establish statistical basis of superheat fuel performance as a function of variables significant to superheat fuel cost.
- d. Perform RML investigations on irradiated fuel to determine design margin.

Nuclear Development Program

1. Pre-Operational Phase

- a. Obtain basic information from irradiation testing to establish limits for nuclear superheat performance.
- b. Do development work on reactor control system for time delay, reliability and application of temperature, power and other control actuation devices.

- c. Establish performance characteristics of equipment unique to superheat system that requires development.
- d. Obtain design information for specific prototype design and investigate methods of improving reliability and performance of the prototype reactor.
- e. Perform analytical study and perform limited experimental program to establish means of improving core performance such as power flattening and temperature flattening.
- f. Perform critical experiment on selected fuel configuration.
Note: This work may not be necessary if results of Base Development are applicable.
- g. Perform core mock-up test and flow distribution tests to determine fabrication techniques, improve performance and reliability, and verify refueling techniques.

2. Post-Operational Phase

- a. Complete start-up test procedure to evaluate all reactor characteristics: criticality control, instrument, scram characteristics, decay heat removal, xenon.
- b. Complete start-up test procedure to evaluate control, coupling and operating characteristics of combined-plant.
- c. Perform rod oscillation tests to measure transient response characteristics.
- d. Determine operational characteristics and design adequacy for reactor cooling, start-up, and decay heat removal systems.
- e. Install orificed fuel elements to evaluate performance of temperature flattening devices.
- f. Investigate power flattening, refueling and control rod patterns to power flatten to increase output.
- g. Check-out operating characteristics of new reactor system instrumentation.
- h. Perform design analysis of prototype first core and extrapolate results for application to second core and larger nuclear superheat reactors.

Superheat Plant Development

1. Pre-Operational Phase

- a. Perform mechanical component development and perform life tests to establish reliability of components such as control drives, vessel nozzles.
- b. Do development work on reactor control system for time delay, reliability and application of temperature, power and other control actuation devices.

2. Post Operational Phase

- a. Evaluate stability, heat transfer and other design characteristics of combined boiler separate superheat system.
- b. Determine reliability, availability and operational characteristics of complete power plant system.
- c. Investigate methods of improving plant performance, reliability and availability.
- d. Develop design criteria for larger or higher performance superheat reactor systems.

Operational Planning

1. Pre-Operational Phase

- a. Determine and specify test programs.
- b. Plan tests and issue specifications.
- c. Develop and specify special instruments for operational tests.

2. Post-Operational Phase

- a. Designate start-up test program.
- b. Coordinate test programs and interpret results.
- c. Perform engineering application studies and make recommendations on superheat power systems.

10.5.5 Prototype Reactor Development Costs

It is estimated that the cost of the development program described in Section 10.5 would be about \$5,722,200 not including the cost of the prototype reactor.

10.6 R&D Cost Estimate for Proposed 300 MW(e) Separate Superheat Reactor Plant

The three phases of the recommended R&D program have been discussed in the previous paragraphs of this Section. The estimated cost of this program is summarized in Table 10.2. The total R&D cost of the program is estimated to be \$20,433,000. Of this amount, approximately \$1,907,354 is funded in the current base development program. In addition, it is currently planned that the \$7,750,000 capital cost of the Hook-on Superheat Reactor at VBWR will be provided by private financing. The remaining development costs would be \$10,776,100. The cost of supporting an alternate development program utilizing a prototype superheat reactor is included in Table 10.2.

The following assumptions have been made in the preparation of the cost estimate.

1. The overhead rates used in the estimate are the approved 1960 provisional rates. No escalation of 1960 dollars was provided.
2. The cost of the Hook-on Superheat Reactor at VBWR and the 75 MW(e) prototype Separate Superheat Reactor are not included.
3. In the development program for the Hook-on Superheat Reactor at VBWR, fuel costs are included for fabrication and cost of conversion of UF_6 to UO_2 but not inventory or reprocessing charges. In addition, the cost of fossil fuel for the auxiliary steam boiler is not included.
4. In the development program for the 75 MW(e) Prototype Separate Superheat Reactor, no fuel charges are included.

TABLE 10.2

R&D PROGRAM COST ESTIMATE

	FY 1961	FY 1962	FY 1963	FY 1964	FY 1965	FY 1966	FY 1967	Total
Nuclear Superheat Project - Base Development Pro- gram - Reference Section 10.3.1	1,632,900	274,454						1,907,354*
Hook-on Superheat Reactor at VBWR - Ref. Section 10.3.2	164,100	695,000	2,064,000	2,006,800	1,610,500	235,700		6,776,100
136 300 MW(e) Separate Superheat Reactor- Ref. Section 10.3.3						1,000,000	3,000,000***	4,000,000
Total	1,797,000	969,454	2,064,000	2,006,800	1,610,500	1,235,700	3,000,000	12,683,454 7,750,000** 20,433,454
Alternate Development Program								
Prototype Superheat Reactor (75 MW(e)) - Ref. Section 10.5	474,400	892,500	851,600	1,283,000	759,000	797,700	664,000	5,722,200

* Funded under Contract AT(04-3)-189

** Currently planned that capital cost of Hook-on Superheat Reactor at VBWR be provided by private financing.

*** Part of these funds would be expended after FY 1967 for post operation R&D.

