GEAP-4105

EVESR NUCLEAR SUPERHEAT FUEL DEVELOPMENT PROJECT

First Quarterly Report, May 4 – August 31, 1962

By
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January 31, 1963
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Atomic Power Equipment Department
General Electric Company
San Jose, California

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GEAP-4105
REACTOR TECHNOLOGY

EVESR-NUCLEAR SUPERHEAT FUEL DEVELOPMENT PROJECT

FIRST QUARTERLY REPORT

MAY 4 - AUGUST 31, 1962

U. S. Atomic Energy Commission

Contract AT(04-3)-189

Project Agreement 29

ATOMIC POWER EQUIPMENT DEPARTMENT

GENERAL ELECTRIC

SAN JOSE, CALIFORNIA

TABLE OF CONTENTS

TABLE OF CONTENTS (CON'D)

TABLE OF CONTENTS (CON'D)

LIST OF ILLUSTRATIONS

LIST OF ILLUSTRATIONS

LIST OF TABLES

1.0 INTRODUCTION AND SUMMARY

1.1 Introduction

This is the first in the series of quarterly progress reports to be issued covering the work performed on the EVESR-AEC Nuclear Superheat Fuel Development Project under Contract AT(04-3)-189, P.A. 29. The EVESR plant design and the design of the first core load of fuel had been in progress prior to AEC participation so that a significant amount of activity had taken place prior to the start of this reporting period. In order to provide information covering this work, a description of the EVESR facility is included along with the technical progress on the AEC-sponsored program.

The EVESR-AEC Nuclear Superheat Fuel Development Project has been organized into the following work task areas:

Sub-task A-1: Program Planning and Evaluation

Sub-task B-1: Mark II Fuel Design

Sub-task B-2: Fuel Design, Engineering Physics

Sub-task C-1: Fuel Process Development and Fabrication

Sub-task C-2: Fuel Element Process Development and Fabrication

Sub-task C-3: Neutron Source Fabrication

Sub-task C-4: Fuel Evaluation Instrumentation

Sub-task C-5: Coolant Chemistry Sampling Equipment (Design & Installation)

Sub-task D-1: Fuel Tests, Engineering Physics

Sub-task D-2: Fuel Tests, Engineering Assistance for Site

Sub-task D-3: Fuel Tests, Test Procedures

Sub-task D-4: Fuel Tests, Data Reduction

Sub-task D-5: Fuel Activity Release and Coolant Chemistry

Sub-task E-1: Post-Irradiation Examination Planning and Evaluation

Sub-task E-2: Pre- and Post-Irradiation Examinations

Sub-task F-1: Advanced Fuel Design and Process Development

Sub-task F-2: Advanced Fuel Fabrication

These tasks will serve as a basic outline for the reporting of progress in this and all future quarterly progress reports.

1.2 Summary

The EVESR reactor is scheduled to go critical sometime in February 1963. Until the reactor becomes operational, the EVESR R&D Program activity will consist primarily of work in the planning, design areas, and fuel fabrication.

Some of the significant results reached to date are as follows:

1. The planning effort has advanced to the point where task reference sheets have been prepared for the entire program. Section 3.0 of this report covers the initial planning activity.

2. The fuel design for the Mark II EVESR core is essentially completed with the exception of certain instrumentation details.

3. Material procurement, tooling, and certain process development work leading up to the final fabrication of the Mark II core has been initiated.

4. The scoping effort on the EVESR R&D instrumentation has been completed and detailed drafting has been initiated. Certain critical items affecting the fuel fabrication and reactor construction schedules are in purchasing, such as thermocouples, thermowells, and flow meters.

5. The requirements for the radiochemistry studies have been scoped and a preliminary rough draft copy has been issued for comments.

6. Work has been started on a data reduction computer program which will efficiently process the large volumes of raw data obtained from the test program.

2.0 EVESR REACTOR AND PLANT DESCRIPTION

2.1 Introduction

Continued success in the development of nuclear superheat is dependent on success in the design, fabrication and irradiation of superheat reactor fuel elements that will demonstrate fuel reliability at high temperature. The development of a reliable nuclear-reactor fuel assembly includes the screening and life testing of numerous conceptual designs over a period of several years. The EVESR reactor is being designed as a flexible fuel test bed with the capability of simultaneously testing a large number of experimental fuel elements in an operating reactor environment. While the need for fuel irradiation capabilities under superheat reactor conditions is the primary purpose for constructing this reactor, it is also expected that information on the operation, maintenance, and operating safety of steam-cooled, power reactor plants will be forthcoming from the operation of the associated power conversion equipment.

In the future it is planned to operate the EVESR reactor and the VBWR as an integrated system. This type of operation will provide an effective means of studying the operation of the separate superheat reactor-boiling water reactor complex under field conditions.

The new fuel test reactor is being erected at the Vallecitos Atomic Laboratory site of the General Electric Company adjoining the Vallecitos Boiling Water Reactor (VBWR)*. The laboratory site is located at Pleasanton, Calif., approximately 25 miles from the headquarters of the Atomic Power Equipment Department of General Electric at San Jose, Calif. The EVESR reactor will add superheat development capabilities to the array of nuclear facilities at the Vallecitos Laboratory which include a radioactive materials laboratory, a critical experiment facility, a nuclear measurements reactor, an engineering test reactor, and a boiling water reactor power plant, as well as laboratories for radiochemistry and metallurgy research.

The site for the new reactor, Figure 2.1, has been selected adjoining the VBWR so as to simplify the connection of the two reactors into a series type of operation with the two reactors supplying steam to the VBWR turbine-generator. Eventual operation in this mode is planned after initial operation of the EVESR reactor as an independent facility has permitted a complete evaluation to be made of the characteristics of the new reactor. The combined operation of the EVESR reactor and the VBWR

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* The Vallecitos Boiling Water Reactor power plant was constructed jointly by the Pacific Gas and Electric Company and General Electric

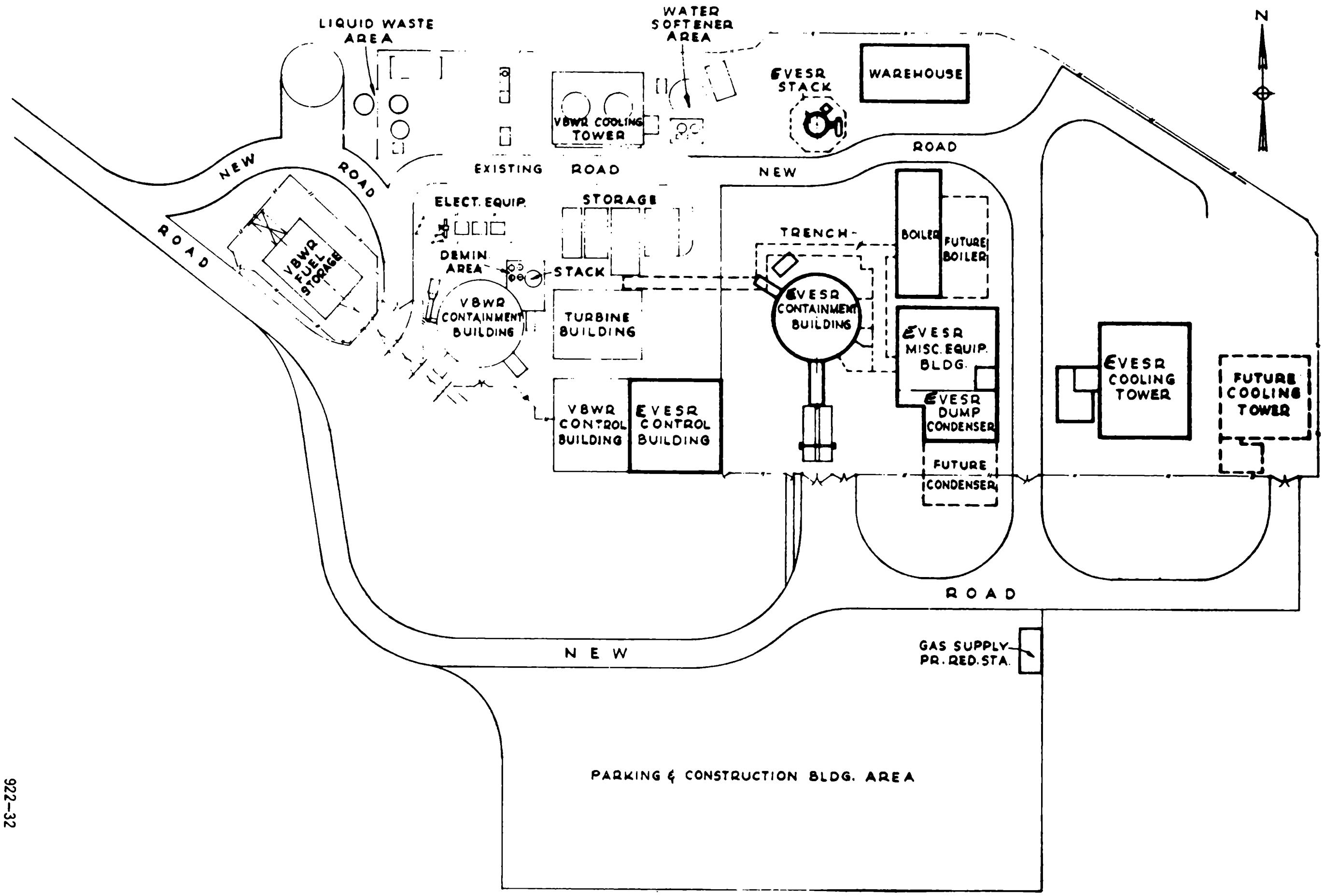


FIGURE 2.1

922-32

will produce information on the performance of separate superheat reactors that will be useful in the design of large nuclear power stations of this type.

The new facility will include the superheat reactor in a conventional reactor containment structure, a gas-fired boiler that will supply saturated steam for cooling the experimental fuel elements, a dump condenser and cooling tower for absorbing the thermal energy generated in the nuclear reactor, an exhaust stack for the controlled release of gaseous wastes, a control and office building and the auxiliary cooling, purification and waste-disposal systems that are required to complete an operating plant. The EVESR reactor will be capable of operation with saturated steam supplied by either the gas-fired boiler, the VBWR, or a combination of both sources of steam. The first operation of the new facility will be at a power level of 12.5 thermal megawatts with saturated steam obtained from the gas-fired boiler. Similar independent operation is possible with the use of saturated steam from the VBWR. A power level of 23 thermal megawatts will be achieved when cooling steam from both the gas-fired boiler and the VBWR will be supplied in parallel to the superheat reactor. These modes of reactor operation will permit steam to be superheated to 1050°F in selected regions of the core. Gross steam temperatures will range from 800 to 900°F dependent upon the particular experimental programs underway at a given time.

2.2 Reactor Description

The EVESR reactor, Figure 2.2, will be a light-water-moderated, thermal-spectrum type of separate superheat reactor containing 32 fuel bundles

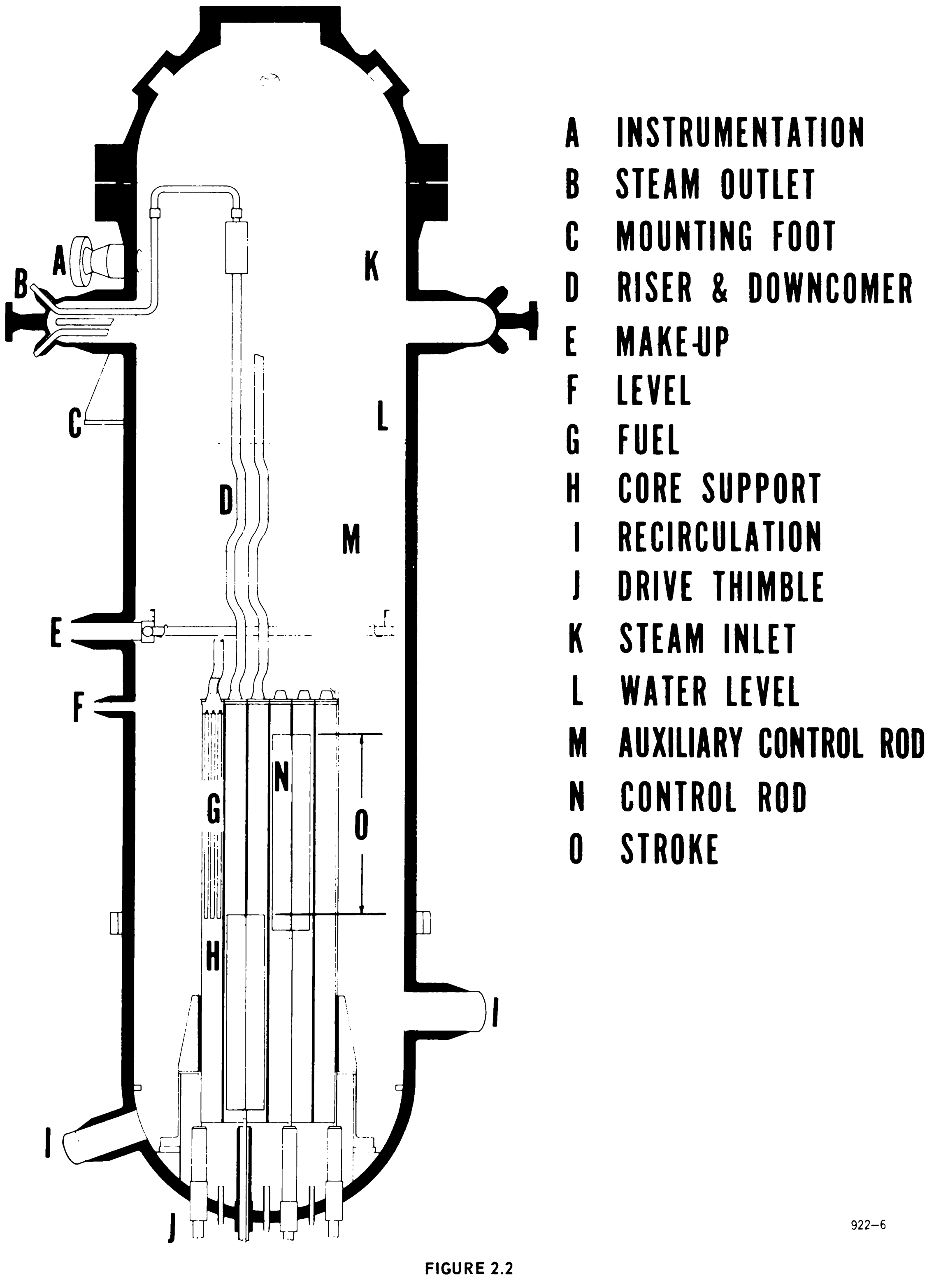

A INSTRUMENTATION
B STEAM OUTLET
C MOUNTING FOOT
D RISER & DOWNCOMER
E MAKE-UP
F LEVEL
G FUEL
H CORE SUPPORT
I RECIRCULATION
J DRIVE THIMBLE
K STEAM INLET
L WATER LEVEL
M AUXILIARY CONTROL ROD
N CONTROL ROD
O STROKE
922-6

FIGURE 2.2

arranged in a rectangular array so as to approximate a right cylinder. Each fuel bundle is individually connected to a superheat-steam outlet nozzle so that there is complete freedom, with the space provided for a single bundle, to remove and replace that fuel bundle with a bundle of different design. The main control rod system utilizes bottom entry rods with bottom-mounted drives so as to minimize the core-support structural problems, reduce the interference of core components during the refueling operation, and simulate the operation of large superheat reactor control systems which, because of power distribution and heat transfer considerations, will probably use bottom-entry control rods.

The reactor pressure vessel has been designed with large head openings to facilitate the refueling operation, and the internal superheat steam piping has been located in the top third of the vessel to allow ready and rapid access for exchanging fuel bundles. The individual superheat steam outlets from the reactor vessel are collected into two pipe headers outside of the vessel so as to simplify the internal piping arrangement and to permit the installation of steam flow control valves in the steam outlet piping. One of the two headers is connected to only four fuel positions, while the remaining header will receive steam from the other 28 fuel bundles.

The four bundle header, known as the "divert" system, is designed to contain steam that has contacted deliberately defected fuel bundles and will guarantee that this steam, that may contain fission gases from the defected fuel, will be carried through shielded pipes directly to the condenser. This feature of the reactor design is provided to assure that the studies of reactor operation with defected fuel may be carried out in complete safety and without interfering with the studies of plant operation with standard fuel. At this time it is not known to what extent the reactor core might be required to expand for experimental reasons, or in fact exactly what type of superheat reactor core might some time be required to be tested. Accordingly, the reactor pressure vessel has been made large enough to accommodate additional superheat bundles or a new core, combining boiling and superheating fuel, such as is the case in an integral superheat reactor.

The reactor pressure vessel design features are shown in Figure 2.3. The vessel was designed and fabricated in conformity with the ASME Boiler Code Section I, and related nuclear code cases, to contain saturated steam at 1250 psig. The vessel was fabricated of carbon steel plate base material clad internally with Type 304 stainless steel. The vessel is similar in many ways to a boiling water reactor vessel, but does have special features to handle superheat steam and to expedite the test operation.

The lower half of the vessel contains nozzles for mounting 24 control rod drives, although initially only 16 positions will be used. Other nozzles in the lower shell and head area include two 10 and two 14-inch nozzles for connection to a future moderator recirculation system, nine thimbles for incore instrumentation, one 2-inch nozzle for attachment to an internal sparger used for both preheating the reactor and the insertion of liquid poison, one 1-1/2-inch drain nozzle, one 6-inch feedwater

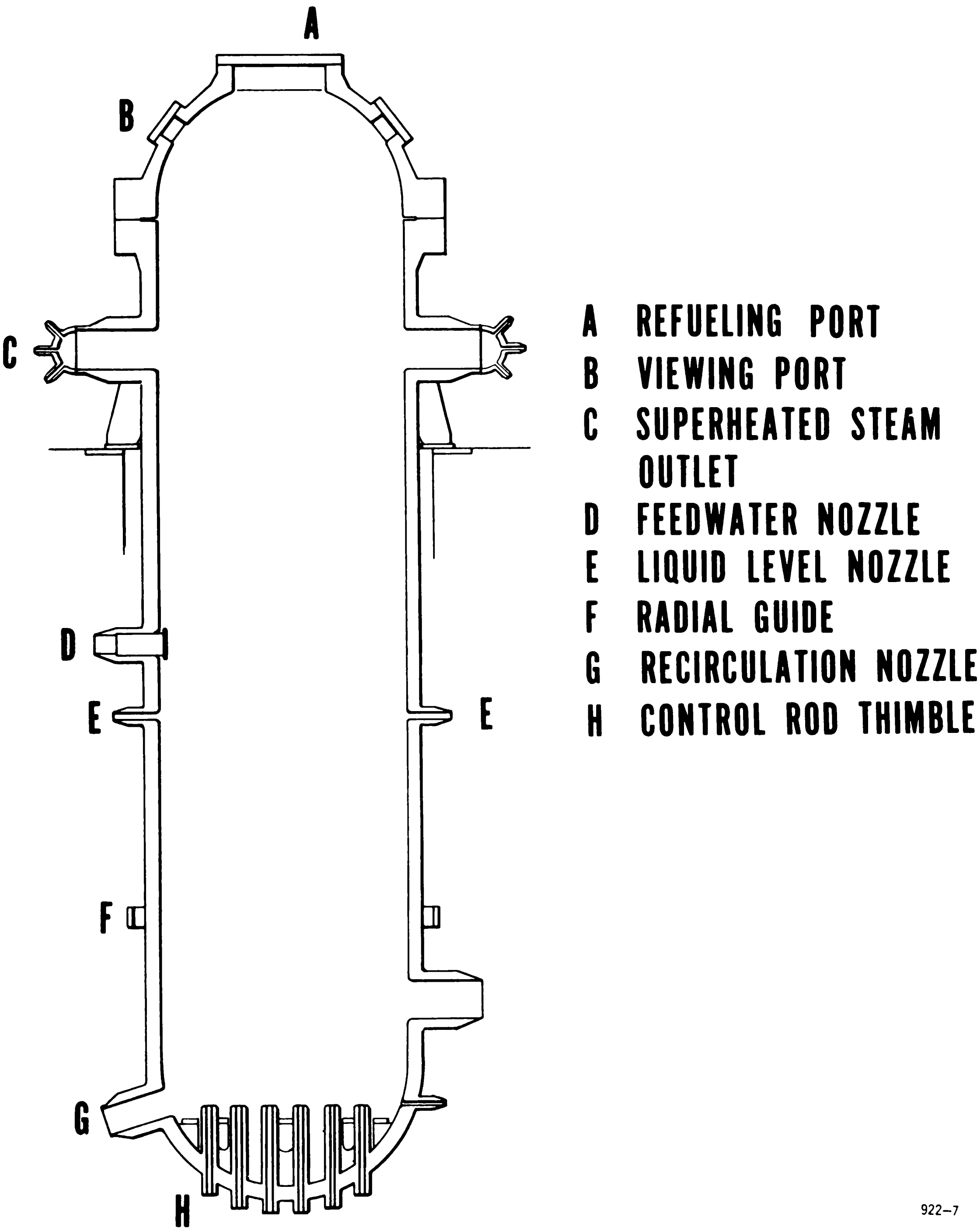

A
B
C
D
E
E
F
G
H
A REFUELING PORT
B VIEWING PORT
C SUPERHEATED STEAM OUTLET
D FEEDWATER NOZZLE
E LIQUID LEVEL NOZZLE
F RADIAL GUIDE
G RECIRCULATION NOZZLE
H CONTROL ROD THIMBLE
922-7

FIGURE 2.3

nozzle with an internal distribution sparger ring and two 3-inch nozzles for liquid level instrumentation.

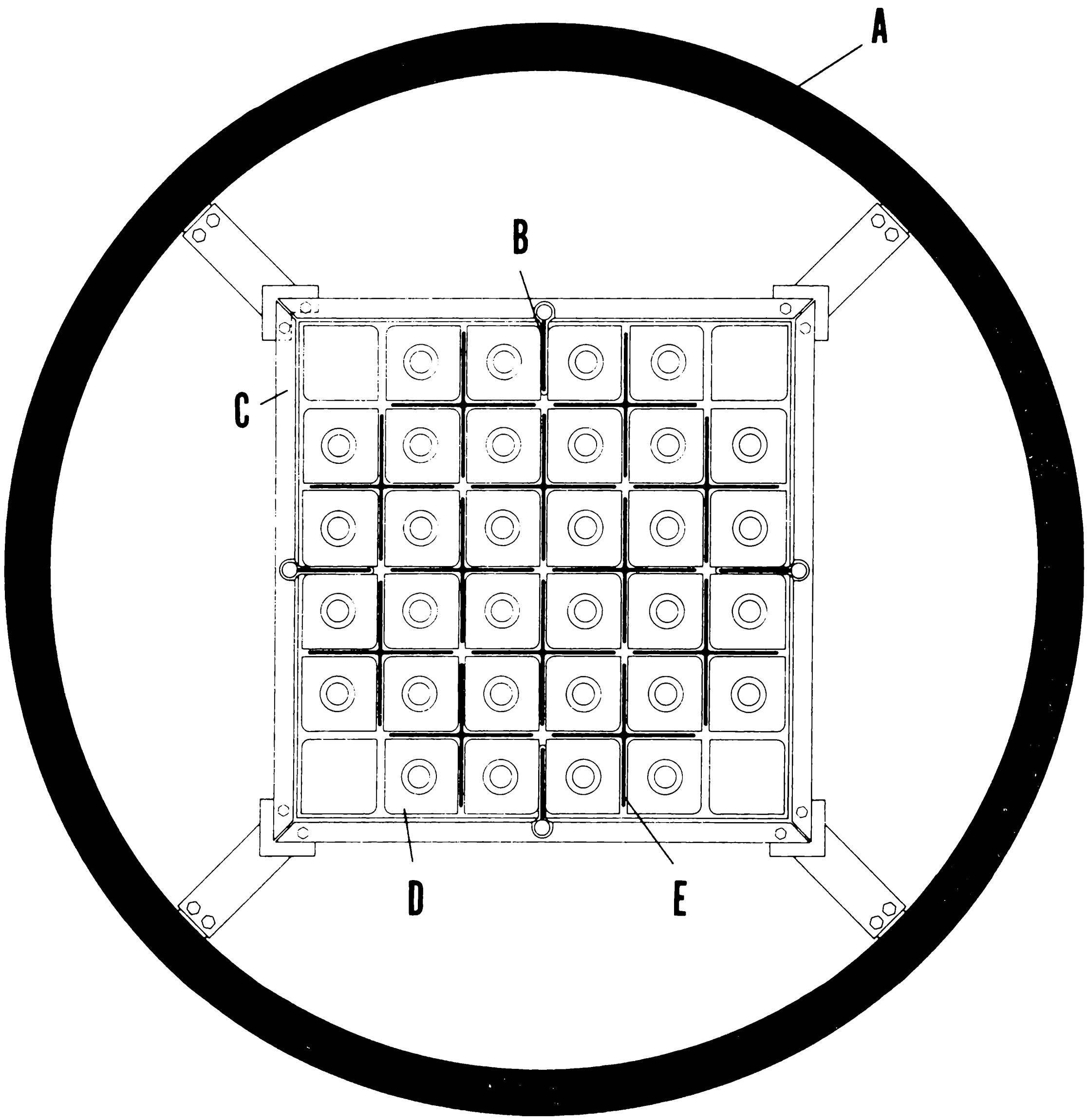
The top half of the reactor vessel contains the special features required for operation of the reactor as a high-temperature test facility. The most interesting feature of the design is concerned with the six 13-1/2-inch dome nozzles into which are installed the superheat steam outlet nozzles. Each dome nozzle consists of a stainless clad sleeve closed by a solid stainless steel dome into which are inserted seven smaller nozzles. A total of thirty-six 2-inch and four 2-1/2-inch nozzles are installed in the six domes, using thermal sleeves to minimize temperature shock to the body of the dome. These nozzles are connected to the internal piping in the vessel that contains superheated steam. Two 3-inch nozzles for connection to safety valves are also installed on the domes. The use of the dome nozzles has increased the effective diameter of the pressure vessel to the diameter that is needed to provide expansion space for the internal superheat steam piping. The effective diameter has been obtained by this method at a cost much less than would have been required to provide a standard vessel of the larger diameter. One pair of 6-inch nozzles in the upper shell region are used to introduce saturated steam into the reactor vessel while a second pair of 6-inch nozzles with flanged connections are provided for the installation of instrument leads that are to be attached to the fuel, or other internal parts of the reactor structure. A single 4-inch and a single 3-inch nozzle used for liquid-level instrumentation complete the list of shell nozzles.

The reactor head is designed to provide the maximum of convenience in the refueling operation. A 31-inch auxiliary opening is provided in the top of the head so that the center 12 fuel bundle positions in the reactor may be reached without the need to remove the entire head. Four 8-inch head openings are also provided for use in viewing the work in the core area and as access ports for the insertion of grapples into the interior of the vessel. In addition to the access openings in the reactor head, provision has been made for the installation of a spray ring for auxiliary cooling of the fuel bundles, and provision has been made for venting the interior of the shell.

Four feet for supporting the vessel within the biological shield have been provided just below the dome nozzles. This support location was selected to minimize the movement of the external superheat steam nozzles and the resultant stress in the superheat steam piping.

The arrangement of the reactor internal components and the bottom-mounted control rod drives is shown in Figures 2.2 and 2.4.

The active portion of the core consists of thirty-two fuel bundles arranged in a square pattern. The four corner positions of the square are not used because of the relatively low power available at those locations. A pattern of twelve cruciform-shaped control rods is spaced so that all fuel bundles except those in the outermost row are surrounded by a control surface on all four sides. The cruciform rods will provide the operating control means for start-up, shutdown, and for the adjustment of power level. Four flatblade auxiliary control rods are located between the two center fuel bundles on each of the four outside rows of the reactor



A VESSEL
B AUXILIARY CONTROL ROD
C SHROUD

D FUEL BUNDLE
E CONTROL ROD

922-8

FIGURE 2.4

core. These blades are used to provide additional shutdown margin over that available in the main control rod system. Because the auxiliary control rods are not used to shim the power of the reactor, they will be positioned either fully inserted, or fully withdrawn from the active core region.

The characteristics of a typical superheater fuel bundle are seen in an examination of Figure 2.5. Nine fuel rods are arranged in a 3 x 3 configuration and are separated from the water moderator by a stainless steel process tube 1.45 inch diameter and 0.025 inch in wall thickness. The active length of the fuel rod is approximately 60 inches. Each fuel rod consists of a stack of cylindrical, sintered UO_2 pellets with a single large concentric hole formed in each pellet. The stack of pellets is clad on the inside and outside with a thin stainless steel tube 0.028 inch thick. The cladding is sealed at each end and is supported from a tube sheet by a tubular extension of the upper end plug. A sealed stainless steel tube is located in the central cavity of each fuel element and serves as a velocity booster to increase the velocity of the steam passing over the inner heat transfer surface of the fuel element. It is expected that a proper matching of the heat transfer coefficients on the inside and outside surfaces of the fuel elements, accomplished by adjusting the flow velocity of the steam by varying the size of the velocity booster to fuel element gap, will minimize the differential expansion between the inner and outer cladding.

The process tube that has been mentioned previously is attached to a second tube sheet in the top of the fuel bundle so that the annular shape

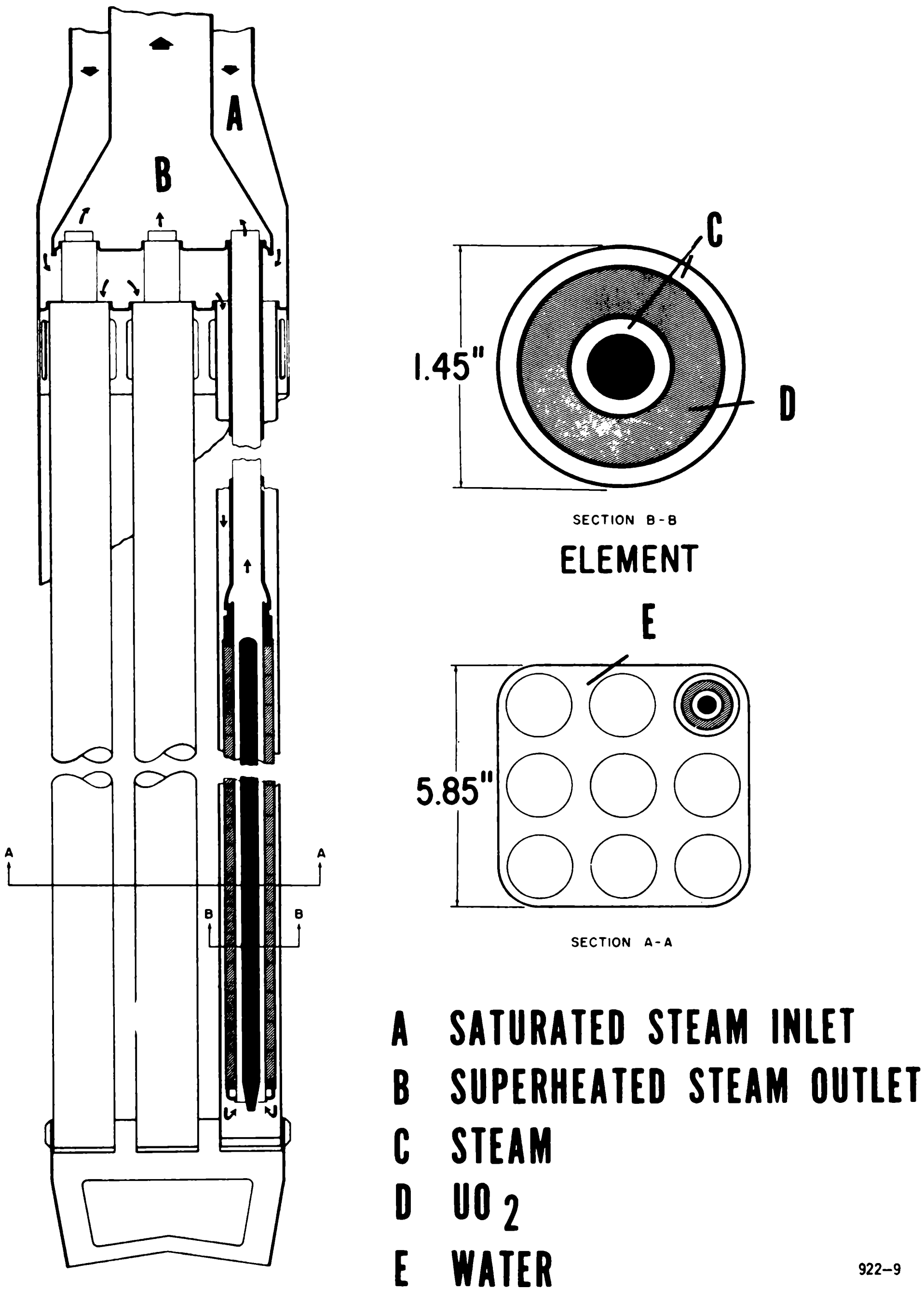

A
B
C
D
E
1.45"
SECTION B-B
ELEMENT
5.85"
SECTION A-A
A SATURATED STEAM INLET
B SUPERHEATED STEAM OUTLET
C STEAM
D UO2
E WATER
922-9

FIGURE 2.5

of the fuel element and the process tube form a two-pass flow system. Saturated steam enters a downcomer pipe above the free surface of the moderator and passes down to the heat generating region of the core. The steam then enters the fuel bundle at a transition piece at the top of the bundle, is partially superheated in passing downward between the process tube and the fuel element outer cladding, makes a 180 degree turn at the bottom of the core and is further superheated in passing upward between the inner cladding of the fuel element and the velocity booster. The superheated steam is collected in a small plenum above the top tube sheet and passes out of the core by means of the internal superheat steam piping that is arranged concentrically with the downcomer pipe.

A system of flanged couplings and rigid pipe connects each superheat steam riser pipe to an outlet steam nozzle. The internal piping has sufficient expansion capability built into each line so that pipe stresses are held within piping-code values.

The array of fuel bundles is fixed in position in the reactor vessel by a stainless steel shroud that surrounds the entire core and is in turn positioned by brackets attached to the reactor vessel shell. An extension to the top of the core shroud provides part of the guide for the auxiliary control blades.

Each fuel bundle is supported by a rectangular zirconium channel that is fixed in position by the adjoining channels and the core shroud. The outsides of the channels serve as guides for the top set of rollers of the cruciform control rods while the inside forms a flow passage for the moderator water that moves by natural circulation past the process tubes.

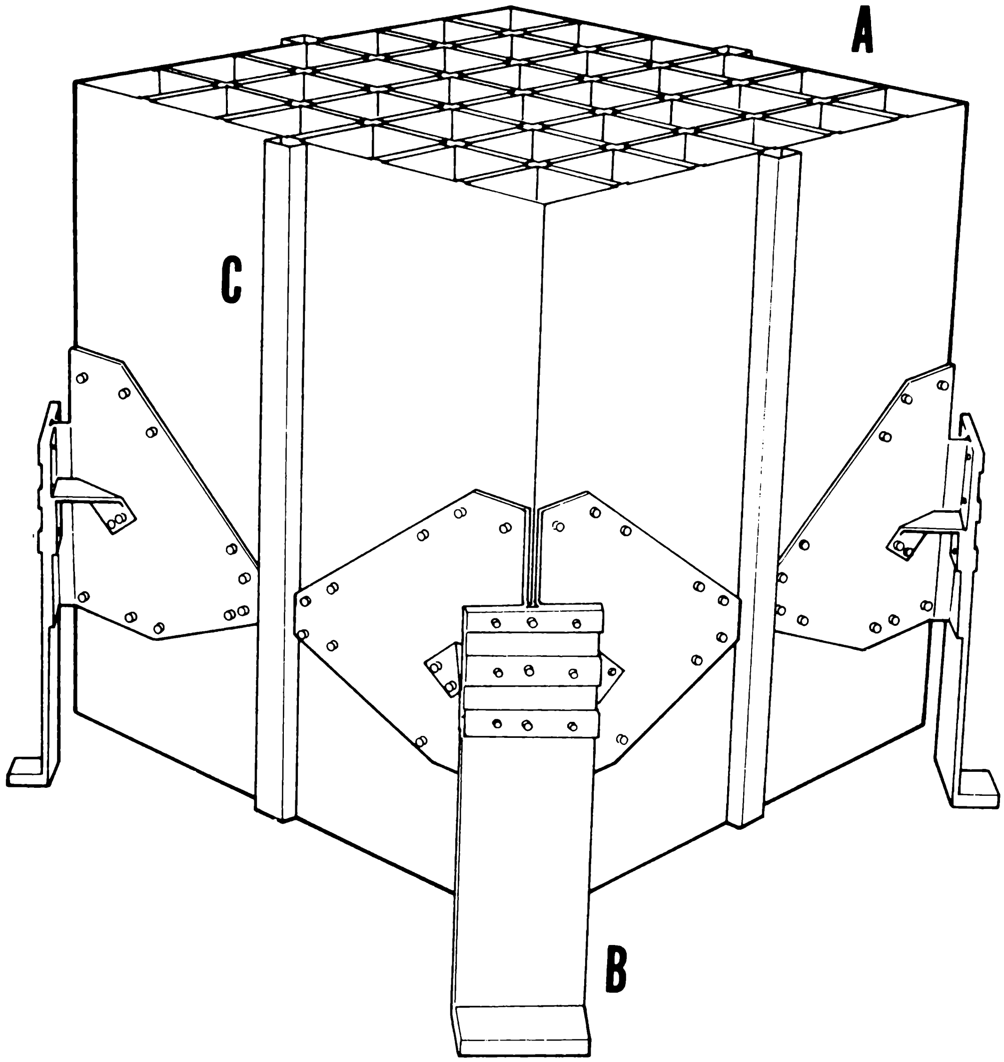
The return path for the flow of moderator is on the outside of the shroud and core support.

The main core support structure for this reactor is shown in Figure 2.6. Instead of the conventional grid plate and supporting angles welded to the reactor-vessel shell, the fuel bundles, channels and core shroud will be supported on a deep rectangular honeycomb built up from bent sections of stainless steel sheet metal that are then pinned together to form the honeycomb shape.

The stainless sheet forming the honeycomb is arranged so that a space is available to guide the lower set of rollers for each control rod as well as to provide a position to locate each of the 36 fuel bundle channels.

The support is completed by a set of four legs fabricated from stainless steel plate. One foot is welded to each leg and in turn each foot is bolted to a corresponding angle bracket welded to the inside surface of the lower head of the reactor vessel. The four legs are bolted to the honeycomb by means of large angle brackets that transmit the core load to the legs without over-stressing the outside surface of the honeycomb.

The radial symmetry of the four core support legs permits the core support to accommodate radial expansion of the core with respect to the reactor vessel without the need for sliding contact of the feet on the support angles attached to the vessel head. The deep plate sections forming the legs provide excellent resistance to lateral bending loads that could



A HONEYCOMB

B LEG

C AUXILIARY CONTROL ROD GUIDE

922-17

FIGURE 2.6

be transmitted to the reactor core by unusual circumstances such as earthquake shock.

The primary control rods will be fabricated in a cruciform shape using Type 304 stainless steel for all parts of the rod. No supplementary poison will be required in these rods as the excess reactivity of the reactor has been maintained at a low value since it is expected that the majority of fuel bundles will be removed from the reactor for metallurgical examination before completing a normal burnup.

The primary control rods are actuated by a twin-screw type of bottom-mounted, bottom-entry, control rod drive that has been developed for use on small boiling water reactors.

The twin-screw, control-rod drive mechanism is attached to the control rod by means of a drive shaft and an extension shaft. The drive shaft penetrates the bottom head of the reactor vessel through a shaft seal which is mounted at the bottom of the vessel thimble. The shaft seal permits linear motion of the drive shaft to move the control rod upward into the core or downward out of the core, and also controls the leakage of water around the shaft by means of a series of close-fitting rings. Demineralized water is injected near the middle of the seal at a pressure above the reactor vessel pressure. Hence, the injection water flows both into the reactor and out to a drain to the reactor water storage tank. The shaft seal is removable while water is in the vessel by engaging a plug on the extension shaft with a fixed seat. The seat is located between the shaft seal and vessel thimble, and upon advancing the drive shaft downward past the normal stroke limit, a seal is accomplished. The drive may be removed separately, if desired, leaving the shaft seal in place.

The auxiliary control blades consist of a black poison, such as boron-carbide powder, encapsulated in sealed stainless steel tubes and sandwiched between thin stainless steel plates. Since these auxiliary control rods will only be used for shutdown of the reactor and will not see a high neutron flux, they are expected to have an extremely long useful life.

2.3 Reactor Safety System

The reactor protection system is designed to protect equipment, plant and personnel by rapidly scramming or running-in the control rods in the event of an accident or the occurrence of a potentially hazardous situation. Control signals which initiate scram originate from a variety of control and detection devices. These signals activate control circuitry to cause the reactor to be immediately shut down and initiate operation of penetration closures, emergency cooling, and whatever other devices are necessary for safe plant shutdown.

The plant conditions which are monitored by the reactor safety system and used to cause scram and other automatic safety measures are as follows:

High Neutron Flux. This condition indicates a reactor power output in excess of the safe level for continuous operation.

Short Period. This indicates an excessive rate of rise of reactor power.

High Reactor Pressure. This control limits the rise in core power due to the small positive pressure coefficient and protects against the need for operation of the reactor safety valves.

Low Reactor Pressure. Possible causes for low reactor pressure include failure of the steam generating source, failure of the pressure regulating valve(s) or rupture of the inlet steam line; all represent loss of cooling steam flow.

High Enclosure Pressure. A pressure rise within the reactor enclosure indicates a major rupture of the high pressure system with possible fuel damage and release of radioactive material. Scramming the reactor minimizes the possibility of release of radioactive material. Isolation of the enclosure is initiated by closing critical enclosure penetrations.

Low Water Level in the Reactor Vessel. This control is used to shut down the reactor before the moderator water level falls below the top of the reactor core and to preserve moderator for flashing to the emergency cooling system to provide the required emergency steam flow through the core.

High Water Level in Reactor Vessel. This control prevents water from rising in the reactor vessel to a point where it may be carried into the superheat fuel elements with subsequent erosion and thermal stresses.

Neutron Flux to Steam Flow Ratio. This control prevents a dangerous rise in fuel temperature due to low coolant flow for the power being generated.

Low Inlet Steam Flow. This control provides a backup to the neutron flux to steam flow ratio scram.

Loss of Plant Auxiliary Power. The reactor safety system, motor generator sets receive their driving power from separate sources in the plant electric system. This control will assure that the safety circuit will function when any electrical power is available in the plant. High moment of inertia is designed into the units so that they will ride through reasonable system disturbances without causing the reactor to scram. For a total loss of electrical power, reactor scram will result because of the fail-safe type of design, after the generator output has decreased below the requirements of the safety system components.

High Seismic Distrubance. This control scrams the reactor in anticipation of possible earthquake damage.

Manual Action. A manual control is readily available to permit the operator to scram the reactor or to manually initiate other safety system functions.

High Radiation Level From Outlet Steam Lines. This control scrams the reactor and isolates the steam lines to minimize fission product transport out of the reactor system in the event of a major fuel failure.

Certain of the safety functions cause automatic rod run-in, which is a slower method of shutting down the reactor, than is a scram. Rod run-in takes 67 seconds for a 50 percent insertion of a control rod while the same insertion by a scram takes place in 0.6 second. The functions which cause run-in do not require the same speed of action as scram, but nevertheless require reactor shutdown because shutdown will be needed in a short time, or because a potentially unsafe condition has been created. If electrical power should be unavailable to run the control rods in, a scram will occur after a 2-second delay. Thus, all run-in signals are actually backed up by a scram.

The reactor safety system is a 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.

2.4 Reactor Instrumentation (Exclusive of R&D)

The EVESR reactor will require extensive instrumentation because of the high-temperature operation associated with the production of superheat steam and also because of the experimental nature of this facility. The instrumentation includes devices for measuring the temperature and flow of the superheat steam as well as for monitoring the moderator level and the reactor pressure. Special thermocouples are provided to detect and indicate the temperature of the reactor vessel at points that are expected to be limiting in the operation of the plant.

Sixteen thermocouples will sense the temperature at critical locations on the outside surface of the reactor vessel. Four of these thermocouples will be selected to be recorded in the control room. The remainder of these thermocouples will be available for reading by means of portable instruments in the control room.

A pressure transmitter senses the vessel pressure and supplies signals for indication and recording in the control room. High and low pressure will be ennunciated from trips on the recorder. The vessel pressure will be indicated within the containment vessel. Four pressure switches sense vessel pressure and supply high pressure trips to de-energize the safety circuit in the event of excessive pressure in the vessel. Four pressure switches sense vessel pressure and supply low pressure trips to de-energize the safety circuit in the event of very low pressure in the vessel.

Two compensating columns, each approximately 180 degrees apart with respect to the vessel, supply water legs for moderator-level measurement.

One liquid-level transmitter senses the liquid level in one of the above columns and supplies signals for an indicating-recording controller in the control room. The moderator make-up flow control valve is regulated by this controller. The control valve is also restricted in its opening rate to prevent excess reactivity changes due to temperature changes. Manual control of the valve will be possible from the control room.

A third compensating column on the reactor vessel supplies a water leg for level measurement from below the moderator make-up inlet to above the top of the head. A liquid transmitter senses the liquid level in this column and supplies signals for a level indicator in the control room.

The temperature of the outlet steam from each fuel bundle will be indicated in the control room. Two thermocouples on each tube are installed up-stream from the steam collection headers. Either thermocouple can be connected quickly to the temperature indicator. Each of the 32 indicators has high and low adjustable trips for alarm.

The 32 individual fuel bundle steam flows are reduced to 12 collection manifold flows. The superheated steam flow will be measured at 18 points and indicated on a local panel within the reactor containment vessel. A low flow trip on any indicator will cause an alarm in the control room.

Thermocouples will be located to sense the bulk outlet steam temperature of each of the two outlet steam lines. Each temperature will be indicated in the control room, and each indicator will be equipped with a high temperature alarm. These temperature measurements will also be used to compensate the bulk outlet steam flow measuring instruments.

Instrumentation will be supplied that will compare neutron flux to inlet steam flow; and, when the ratio of normalized flux to normalized flow exceeds about 1.25, a trip will de-energize the safety circuit. When this same ratio exceeds about 1.35, a second trip will initiate emergency cooling. Two flux amplifiers will receive flux level signals from the two log-N channel compensated ion chambers, automatically, as soon as the period scram function of these instruments has been bypassed (above about 1 percent of rated power). The signals from the flux amplifiers will be compared with the signals from two flow measuring instruments, providing two flux/flow signals for indication, recording, trip and alarm purposes.

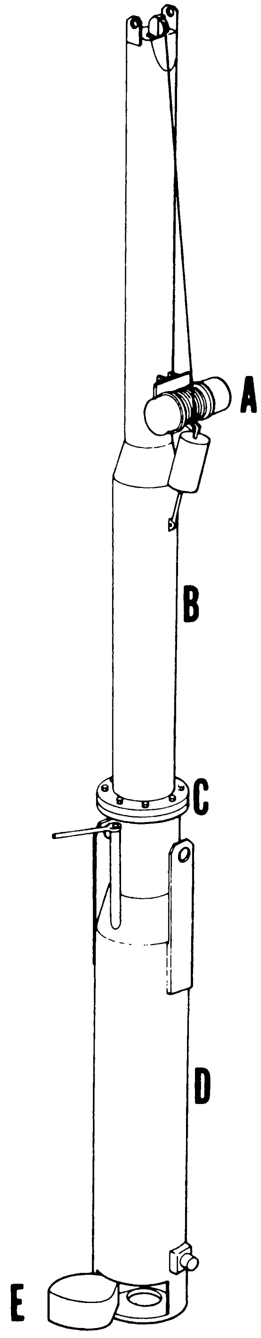
An instrumentation system is provided to monitor the neutron level and control the reactor from start-up through full power. The instrumentation covers the range in three phases; i.e., start-up, intermediate range, and flux level or power range.

2.5 Refueling System

It is planned to refuel the EVESR reactor by the use of a fuel cask and a refueling pool located within the containment structure. The cask method of refueling has been selected because it represents the minimum cost system that will still permit manual manipulation of core components within the limited confines of the containment building.

In order to minimize the developmental problems of the superheated steam piping system within the reactor vessel, the first core for the EVESR reactor will be fabricated with the steam riser and downcomer piping permanently attached to the fuel bundles. This decision has made it necessary to remove a much longer fuel assembly from the reactor than will be expected to be handled in a commercial superheat reactor of this type. The fuel cask that has been designed for this purpose will handle the full-length fuel bundle and riser/downcomer piping for both transfer of the assembly from the reactor to the refueling pool and for transfer of spent fuel from the containment building to an outside laboratory for examination.

The refueling cask concept is shown in Figure 2.7. So as to make maximum use of the cask, it has been designed in two sections, separable at the center. When the two sections are fastened together, the cask is suitable



A ELECTRIC HOIST

B RISER SECTION

C SEPARATION FLANGE

D FUEL SECTION

E DOOR

922–18

FIGURE 2.7

for handling the full-length assembly. When the top section of the cask has been removed, the bottom portion may be used for the transportation of the active portions of the fuel assembly or radioactive core components such as control rods.

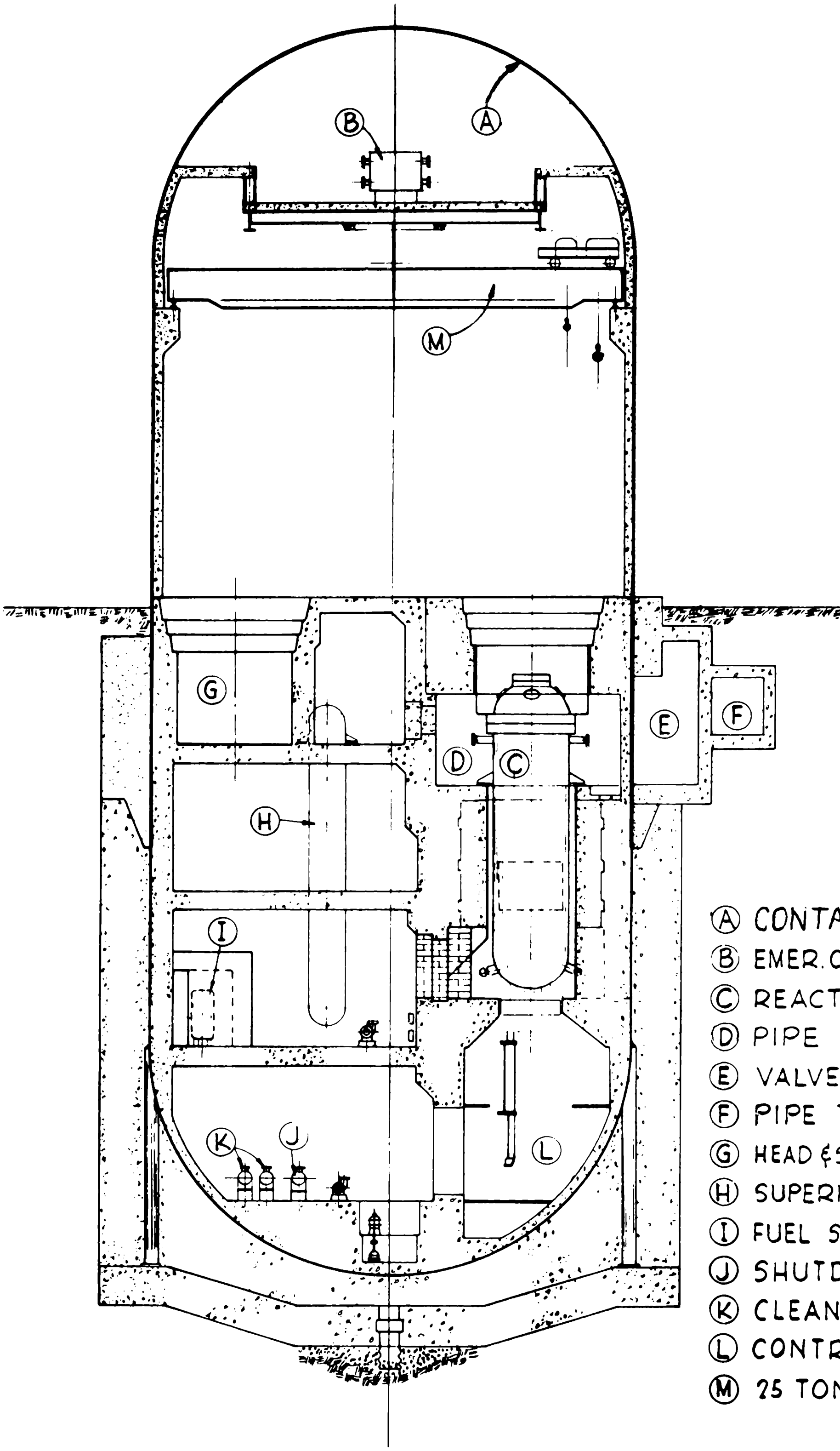
An external electric hoist has been provided as part of the cask and works with an internal grapple to lower a fuel bundle into the reactor core, or to remove the active fuel bundle from the same region. While the refueling cask is suitable for handling either new or irradiated fuel bundles, new fuel may also be lowered into the reactor by the use of the overhead crane that is part of the permanent building structure.

When the cask is used in a refueling operation it is supported on a temporary structure that is placed in the cavity directly above the reactor head opening. This structure supports the weight of the cask and prevents any possibility of overloading the reactor vessel flange or support structure, which could be the case were the heavy cask rested on the vessel head during the refueling operation.

A four-wheel dolly is available for transporting the refueling cask through the equipment air lock in the event that it is desired to remove the cask from the reactor enclosure.

2.6 Building Design and Equipment Placement

The containment building is a conventional capsule-shaped vessel, consisting of a 48-foot diameter cylindrical section, and two hemispherical heads, having a total height of 128 feet, as can be seen in Figure 2.8.



Ⓐ CONTAINMENT SHELL
Ⓑ EMER. COOLING HEAT EXCHANGER
Ⓒ REACTOR VESSEL
Ⓓ PIPE GALLERY
Ⓔ VALVE PIT
Ⓕ PIPE TRENCH
Ⓖ HEAD & SHIELDING PLUG STORAGE
Ⓗ SUPERHEAT STEAM DUMP TANK
Ⓘ FUEL STORAGE POOL FILTER
Ⓙ SHUTDOWN HEAT EXCHANGER
Ⓚ CLEANUP HEAT EXCHANGER
Ⓛ CONTROL ROD ROOM
Ⓜ 25 TON POLAR CRANE

FIGURE 2.8

922–25

Equal heights of the building are above and below grade. The plate thickness of the cylindrical section is 1.015 inch, while that of the two heads is 0.513 inch. The building will be designed as an ASME code vessel, designed for 58 psig internal pressure with a coincident temperature rise of 250°F.

As shown in Figure 2.9, two entrance locks are provided giving access to the operating floor at approximately 120 degrees to each other. These entrances will be a 7-foot, 7-inch, diameter equipment and personnel lock, and a 2-foot, 6-inch, diameter personnel escape lock. Doors on both locks will be interlocked, with the larger lock door being both manually or hydraulically operated.

On the north side of the containment building is the fuel storage pool. The top of the pool is at the operating floor level, and it extends downward for a depth of 27 feet. The pool is constructed of aluminum alloy, and is capable of storing 73 fuel elements and 12 control rods. When filled to its normal level, the pool will contain 45,000 gallons of demineralized water.

The dump condenser building shown in Figure 2.10 is essentially two buildings with a common wall between them. The southern portion contains the condenser, condensate pumps, and air ejectors. These units will have some radioactivity associated with their service and must be shielded. The shield walls are constructed of 24-inch thick concrete.

The present VBWR office building and control room will be extended to the east and will house the EVESR reactor control room on the second floor.

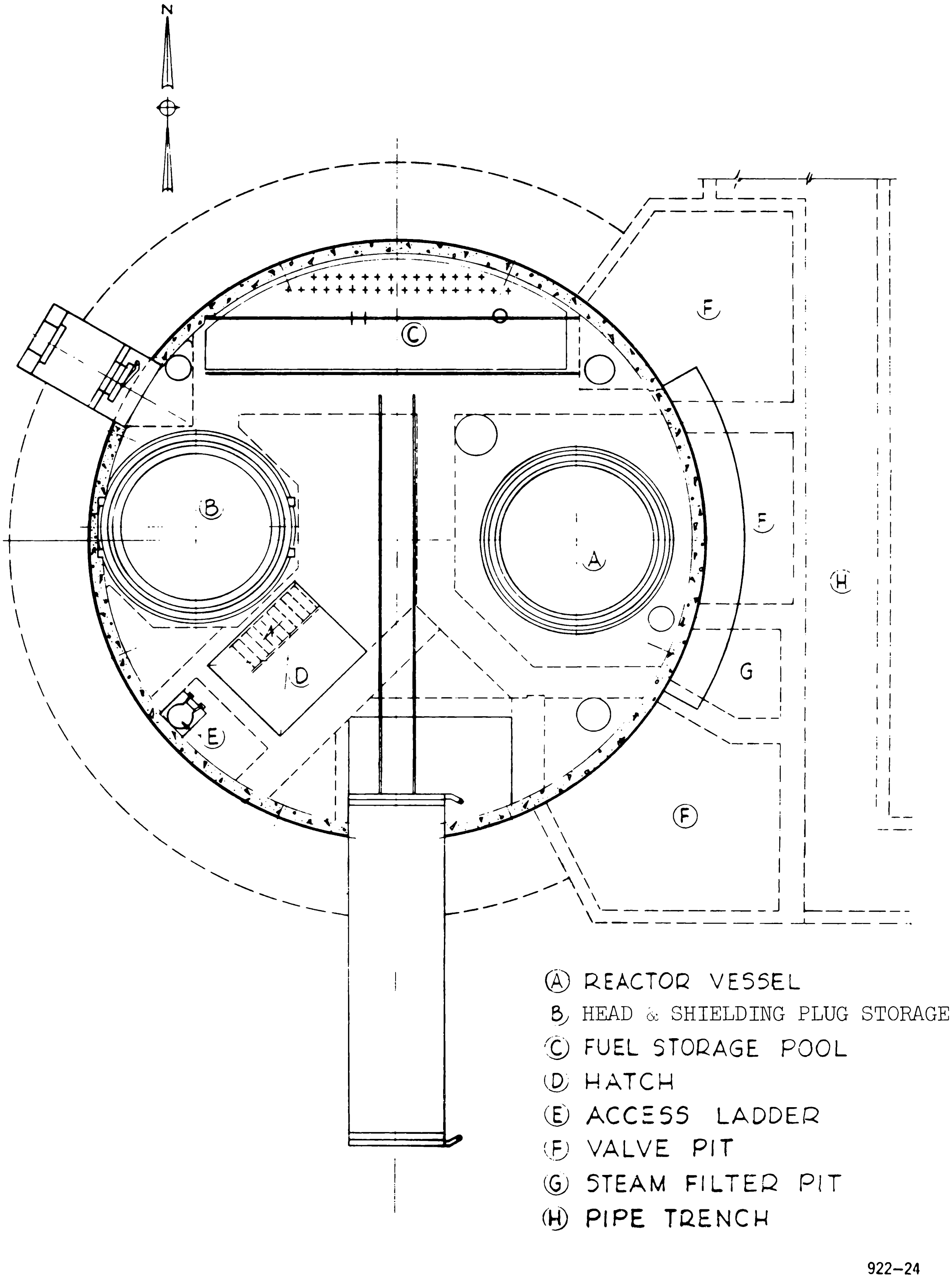
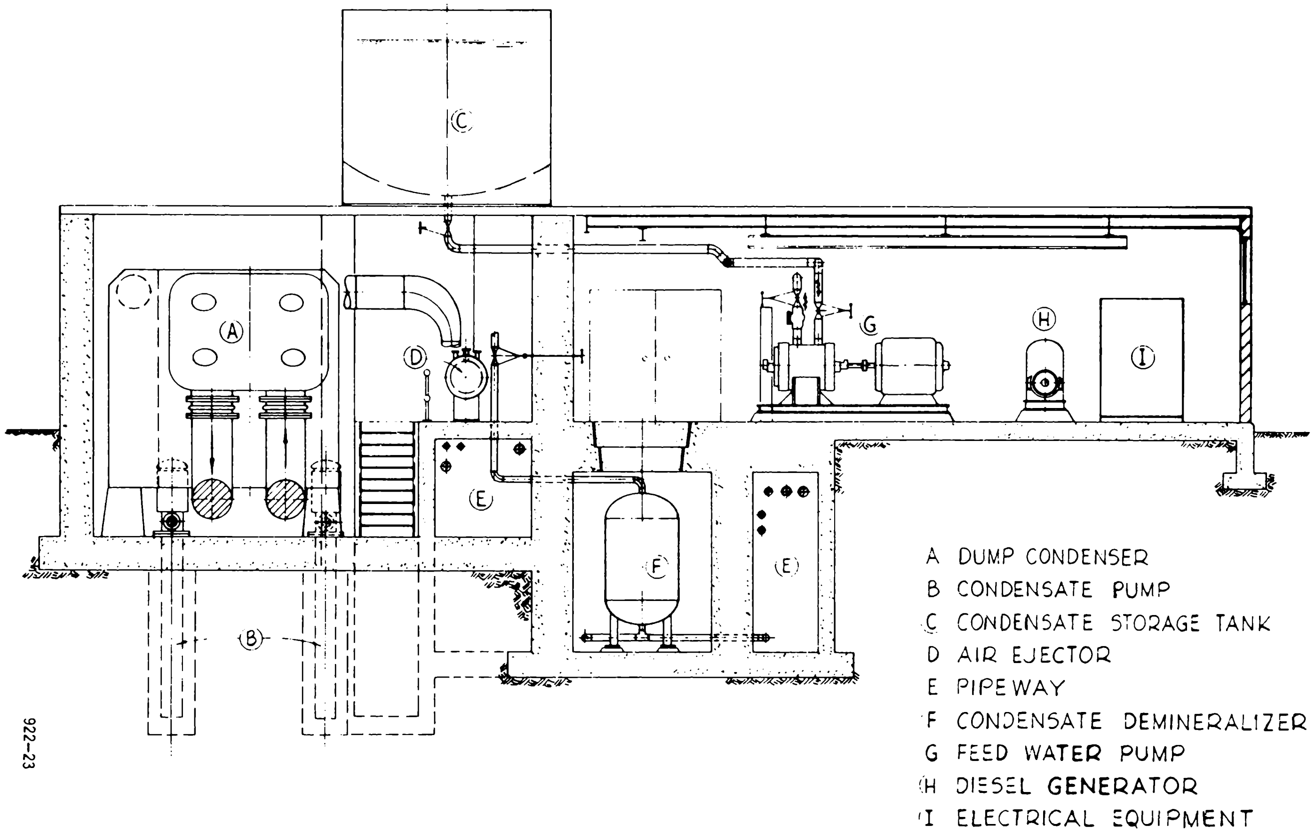

N
C
B
A
F
F
H
G
D
E
F
(A) REACTOR VESSEL
(B) HEAD & SHIELDING PLUG STORAGE
(C) FUEL STORAGE POOL
(D) HATCH
(E) ACCESS LADDER
(F) VALVE PIT
(G) STEAM FILTER PIT
(H) PIPE TRENCH
922-24

FIGURE 2.9



922-23

FIGURE 2.10

The first floor will contain lockers, showers and clean and contaminated clothing storage rooms. Also located on this floor are additional offices, the relay room, and boiler room. An observation area will provide space for visitors to view both the EVESR reactor and VBWR control room operation.

2.7 Steam Cycle

As has been mentioned previously, the EVESR reactor has two sources of 1000-psig saturated steam. These sources are the VBWR and an auxiliary gas-fired steam boiler. The boiler is of conventional, fully automatic, watertube, steel-cased, outdoor design. The saturated steam, from either or both sources will enter the containment building through a single pipeline which then splits and enters the reactor vessel through two nozzles. At this point, the saturated steam flows into and through the superheat process tubes where it cools the fuel and becomes superheated. It then passes out of the vessel through individual fuel bundle nozzles to the superheat steam collection and headering system.

By examining Figure 2.11, it can be seen that there are two heat sinks, the VBWR condenser, and the EVESR reactor dump condenser. An alternate, and parallel flow path to the VBWR condenser is provided through the VBWR turbine. In all cases, the superheat steam is reduced in pressure and temperature by reducing stations and desuperheaters.

The new condenser is a horizontal, two-pass, 14,000 square foot surface condenser designed to operate at 6-inch Hg abs. The hotwell is designed to provide 5 minutes of definite retention to allow radioactive gas decay time. The deaerating section of the condenser will limit the maximum oxygen content of the condensate to 0.01 cc/liter.

The VBWR condenser is similar to the new condenser, but is only about two-thirds of the size of the new unit; however, it also will operate at about 6-inch Hg abs. The two condensing systems are interconnected to equalize flow discrepancies through the piping and the condensate storage tank.

2.8 Condensate and Feedwater Cycle

The VBWR and the EVESR reactor facilities are arranged with similar but separate condensate-feedwater cycles. At both hotwells, high level and low level control valves maintain the hotwell level between predetermined points by taking suction from, and by discharging water to the EVESR reactor condensate storage tank. This interconnection, Figure 2.11, allows the one storage tank to feed both systems. The condensate from the VBWR condenser will be pumped through its existing circuit with any excess or shortage of flow supplied to or from the storage tank. The condensate from the EVESR reactor condenser will be pumped by two 60 percent capacity pumps, through one of two full-flow filters, demineralizers, and post filters. At the outlet of the post filters, the flow is divided into three parts. The largest amount goes directly to the suction of the feedwater pumps. The remaining flow is divided between the shield cooling coils, and the desuperheating water for the steam flow to both condensers. Any excess or shortage will be either discharged to, or made up from the EVESR reactor condensate storage tank. Additional make-up for either system will be provided from the existing 500,000-gallon raw-water storage tank through filters and demineralizers.

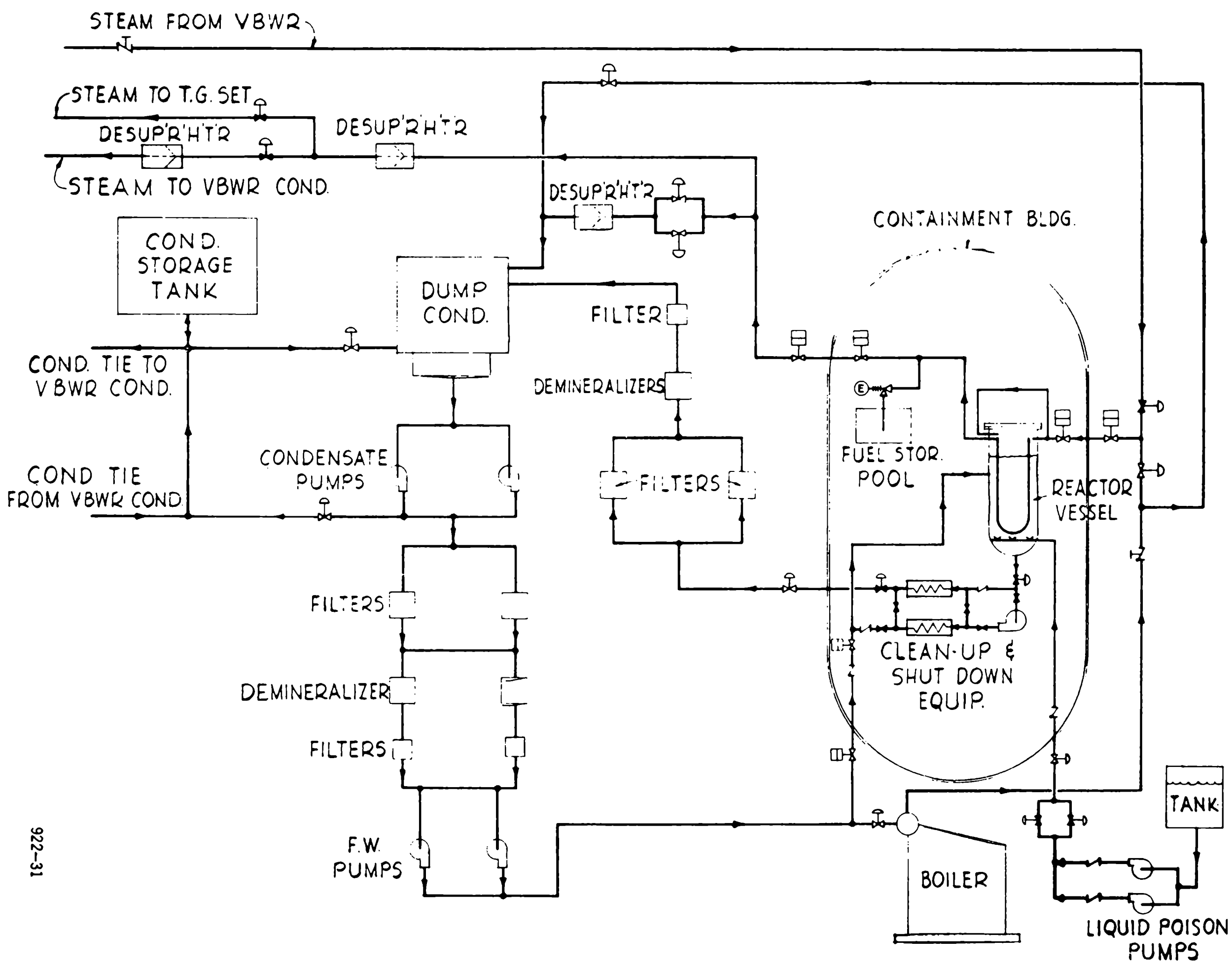
The discharge water from the feedwater pumps is divided into four parts. The majority of the flow will go directly to the gas-fired boiler, through a three-element feedwater control valve. The remainder is divided among the reactor vessel moderator make-up through a single element level control valve; the control rod drive water seals through a pressure regular; and the desuperheating water for the superheat steam to the VBWR turbine-generator.

The superheat reactor moderator water is cleaned during operation by flowing 6 gpm through a cleanup heat exchanger to reduce the water temperature, then through a pressure reducing valve, to reduce the pressure to 80 psig, and finally through one of two full-flow cleanup filters, a demineralizer, and a post filter to the dump condenser. Only one demineralizer and post filter is needed in this circuit, as the cleanup loop can be out of service long enough to change the resin and filter cartridge. When the reactor is shut down, valving is changed allowing a pump to circulate the water through the cleanup system.

2.9 Emergency Systems

An emergency cooling system has been provided for the EVESR reactor to assure protection against loss of cooling steam flow through the fuel passages. Since the emergency system must be capable of operation when the containment building has been isolated from all outside sources of steam or water, a system was devised that makes use of the saturated water in the reactor vessel as a source of cooling steam. During operation, the reactor vessel will contain a minimum of 700 cubic feet of saturated water at a

FIGURE 2.11



922-31

pressure of nearly 1000 psig. Any reduction in vessel pressure will result in a flashing of a portion of the moderator until an equilibrium pressure is again achieved. This natural phenomenon is used to achieve the steam flow that is used to cool the superheat reactor fuel following a scram and loss of normal steam flow. When a scram, followed by isolation, has occurred the reactor pressure is reduced at a controlled rate so that the required steam flow is established without at the same time reducing the pressure at too rapid a rate and thereby wasting the available steam. An energency condenser is provided within the containment building. This condenser serves to return the moderator water to the reactor vessel and maintains the moderator volume so that eventually a long-time cooling cycle similar to that used in boiling water reactors may be established. After the fuel element heat generation rate has reduced to a level suitable for radiation cooling through the steam gap to the moderator, the steam flow requirements for direct cooling of the fuel element will be negligible and a natural convection cooling cycle will be adequate to hold the fuel temperature to a safe value.

A liquid poison system is incorporated in the design as a final back-up for the reactor shutdown systems. A solution of sodium pentaborate dissolved in water, which can be pumped into the reactor by high pressure pumps in ten minutes or less, is sufficient for reactor shutdown. The pump and valve controls, which must be manually initiated, are two in-series for protection against inadvertent operation.

In case of normal power failure the reactor safety systems will place the reactor in a safe condition. The reactor would be scrammed and the emergency cooling would prevent the fuel from overheating. Emergency power would be available from a 75 KW diesel generator.

2.10 Circulating Cooling Water System

The new cooling water system will be kept independent of the VBWR system. A new cooling tower of induced draft design is used for removing heat from the condenser circulating water. Tower duty is 266,000,000 Btu/hr with a flow of 14,000 gpm, an ambient wet-bulb temperature of 67°F, and a water temperature decrease of 38°F. The cooling tower foundation forms a basin and pit from which the circulating water pump and the cooling water pump take their suctions.

2.11 Building Ventilation

Conditioned air is supplied to the containment building by means of a centrifugal-type fan which drains outside air through an evaporative cooler and gas-fired furnace at the rate of 15,000 cfm. This air flow represents about six changes per hour for the net building volume. The entire volume of air enters the containment building through a single 24 inch O.D. penetration located approximately 12 feet above the operating floor. Two isolation valves, one inside and one outside of the containment shell, are installed at the penetration.

Fresh air is distributed to the dome, and to each of the various operating levels through a system of ducts and volume control dampers.

Approximately one fifth of the total supply volume is passed through the control rod room and upward through the annular space between the reactor vessel wall and the biological shield wall. All exhaust air is collected in the pipe gallery where it is carried to the stack in a buried exhaust line. A fan, installed above grade at the stack, discharges the air into the stack, about 10-1/2 feet above grade.

Noncondensable gases, composed of oxygen, hydrogen, air, and radioactive gases are released from the VBWR and EVESR reactor-condenser air ejectors into the stack at a point 18 feet above grade, which is 7-1/2 feet above the fan breeching. The off-gas streams from both condenser air ejectors are routed independently to the new stack. Following the air-ejector condensers, the gases are preheated by the use of electric heaters to reduce the relative humidity before passing through an activated charcoal iodine absorber. At normal gas flow, the piping is sized to provide approximately a 30-minute holdup delay time. After the holdup pipe, the gas streams pass through high efficiency filters prior to stack release. Each of the air-ejector, off-gas streams is provided with monitoring equipment to determine the level of any radioactive materials. This equipment will automatically close the isolation valves if excess activity levels are detected.

The new ventilation stack is an unlined concrete stack 160 feet high, located about 100 feet north of the EVESR reactor containment building. The exit velocity is approximately 50 fps. A monitoring system is provided, with the sampling point about 70 feet above grade.

2.12 Waste Disposal System

The EVESR reactor liquid waste disposal system will be connected to the existing VBWR system. Pumps will be installed to transfer the water from various sumps, containment building, valve pit, pipe trench, boiler, and the feed storage pool, to the existing VBWR equipment. Liquid waste will be first directed to either or both existing 5000-gallon waste water storage tanks where the liquid may be stored for a decay period. It is then demineralized for reuse, or pumped to the 60,000-gallon waste disposal tank. From this point it will either be pumped to the existing VBWR 4000-gallon condensate storage tank, or the new 10,000-gallon condensate storage tank. The existing 60,000-gallon waste disposal tank is provided with a recycle line and filter for reducing the radioactivity of the waste if required, to a level permissible for truck disposal.

Solid wastes will generally be disposed of by using the methods, procedures, and controls established at VBWR. These wastes will also be taken to an existing area, presently used by VBWR, and removed by a licensed agency for off-site disposal.

2.13 Shielding

The basis of design of the containment building is such as to allow continuous access to the operating floor during full-power operation of the reactor. The permissible dosage received by any of the plant personnel is considered to be 100 mr/wk. The design limit on the operating floor is 0.5 mr/hr. Below the operating floor, larger rates are acceptable, allowing only limited access to most of these areas.

The operating floor area adjacent to the reactor will be cooled by circulating condensate through pipes imbedded in the concrete. This cooling will maintain the concrete temperature at something less than 200°F.

Shielding of the steam lines is provided in two areas. The first, is the pipe gallery around the top portion of the vessel. The second is the area where the pipes run underground. The level of activity on the operating floor, and at ground level over the pipe trench must be maintained at 0.5 mr/hr or lower. Not only is potential iodine and nitrogen activity in the superheat steam to be considered, but also the activity in the saturated steam from VBWR. The concrete operating floor over the pipe gallery is over 9 feet thick, while the covering of the pipe trench must be equivalent to 5 feet of earth plus 1 foot of concrete.

The demineralizers and filters are all located such that they are able to be operated when the plant is at full power. The operation of valves, sluicing out of spent resins, recharging of new resins, and changing the filter cartridges is performed in areas behind concrete walls, where the activity level is designed to be 0.5 mr/hr.

The control rod drive area will be inaccessible during reactor operation. However, after reactor shutdown, this area will be low enough in activity level to allow plant personnel to enter for maintenance and testing.

3.0 SUB-TASK A-1 PROGRAM PLANNING AND EVALUATION

3.1 Program Organization

The entire EVESR Nuclear Superheat Fuel Development Project has been analyzed and organized into task assignments which can be delegated to responsible performing components within the Atomic Power Equipment Department. Task reference sheets have been prepared for each of these assignments. Presented in Appendix I is a detailed listing of each development task showing objectives, approach to problem, expected results, and a bar chart schedule of significant events. Work has been initiated on a number of these tasks and the progress in each is given in subsequent sections of this report. As the program progresses, it may be necessary to add to or delete from this basic list of tasks as the development emphasis shifts.

4.0 SUB-TASK B-1 MARK II FUEL DESIGN

4.1 Historical Background

The Mark I initial EVESR core was designed during the period January 1960 to November 1961 and consisted of a full core loading of uniform enrichment UO_2 encased in Type 304 stainless steel cladding material. Very reliable driver-type fuel performance was expected from this fuel until subsequent test results from the SADE fuel test facility indicated a serious potential fuel cladding problem associated with stress-corrosion cracking of Type 304 stainless steel fuel claddings.

In November, 1961, the seriousness of the EVESR cladding problem was recognized and a new core design was initiated consisting of eight different clad materials, three different UO_2 enrichments, power flattening shims, and extensive instrumentation. This new fuel has been designated as Mark II fuel. The core composition presently planned is as follows:

No. Bundles	Cladding Material
8	304 DVM Stainless
8	Inconel
8	Incoloy
4	310 Stainless
1	304 Commercial Stainless
1	316 Stainless
1	347 Stainless
1	348 Stainless

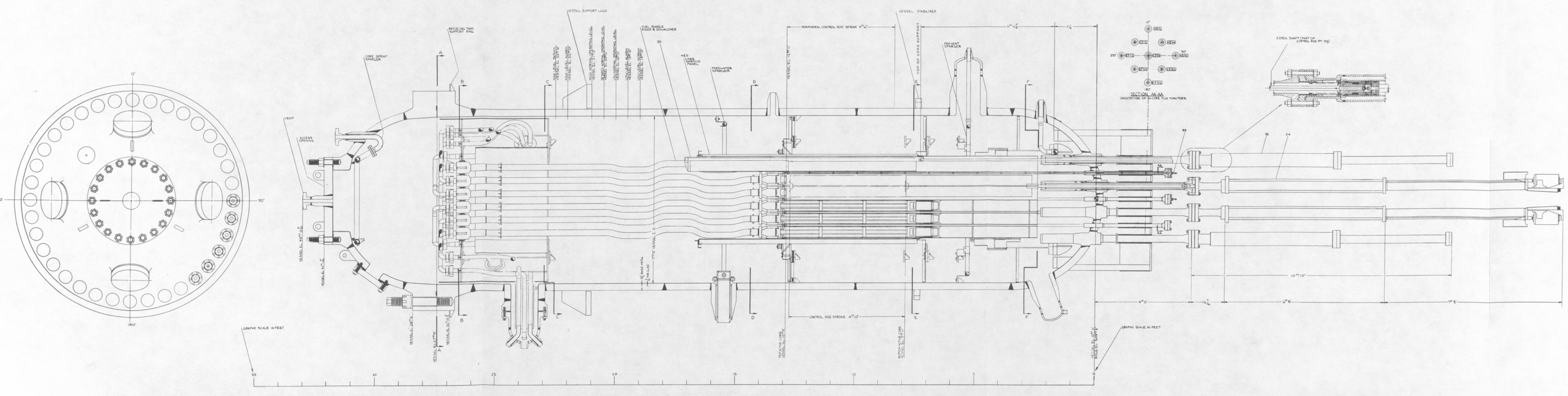
The design and drafting associated with the fuel design is being handled in three phases. Phase I will be the issuance of the drawings for 28 non-instrumented fuel assemblies. Phase II consists of those drawings detailing the addition of instrumentation to eight of the Mark II fuel assemblies. Phase III covers the details of a fully-instrumented fuel assembly. The details of the instrumentation installation in the fuel assemblies is presently in the conceptual stage. The type of instrumentation to be installed is covered in Task C-4.

4.2 Current Design Effort

The proposed arrangement of the Mark II fuel bundles with respect to the reactor vessel and the other internal parts of the reactor is shown in the reactor assembly drawing, Figure 4.1. The specific location of individual fuel bundles in the overall core plan, the cladding material used in each individual fuel bundle and the instrumentation installed on the fuel bundles is shown in Figure 4.2..

The mechanical details of the Mark II fuel bundle, the first type of fuel bundle for use in the EVESR, are shown in Figure 4.3. Inasmuch as the EVESR is designed for use as a fuel irradiation test facility, certain features are incorporated in the bundle design to simplify the fuel handling operation and to assure the maximum reliability of the mechanical design. For example, both the steam entrance opening and the exit connections to the fuel bundle are made at the top of the bundle to eliminate the need for flanged joints below the normal water level in the reactor vessel. For the same reason, the fuel bundle is designed with all-welded construction.

The EVESR fuel bundle, designated as the "Mark II" type for ease of identification, consists of two major sub-assemblies: the "riser-downcomer"assembly which connects the fuel bearing region of the fuel bundle to the internal piping in the reactor vessel, and the cluster of fuel elements forming the active region of the fuel bundle.



PART NUMBERS APPLY TO THIS DRAWING AND TO THE SUCCEEDING SECTIONS

PART NO.	NAME
1	FUEL CHANNEL G1
2	FUEL CHANNEL G2
3	CORE SUPPORT & MOUNT
4	SHROUD PANEL G1
5	SHROUD PANEL G2
6	ANGLE RING
7	HOLDER
8	COVER
9	BRACKET
10	ANGLE
11	BARRIER
12	NUT PLATE
13	NEUTRON WINDOW
14	TEE RING G1
15	TEE RING G2
16	TEE RING G3
17	TEE RING G4
18	CONTROL ROD
19	NOT USED
20	CHANNEL
21	NOT USED
22	NOT USED
23	NOT USED
24	ANGLE – SHROUD
25	SHIM
26	STUD
27	PIN
28	RIVET
29	NOT USED
30	DOWEL
31	NOT USED
32	BOLT
33	NOT USED
34	CONTROL ROD DRIVE
35	AUXILIARY SCRAM ROD
36	6 IN. INSTRUMENT GUIDE TUBE
37	SCREW
38	STEAM DIFFUSER COUPLING
39	STEAM DIFFUSER
40	IN-CORE MONITOR GUIDE TUBE G1 (MONITORS AB12, AB56, CD34, EF12, AND EF56)

PART NO.	NAME
41	IN-CORE MONITOR GUIDE-TUBE G2 (MONITORS AB34, CD12, CD56, AND EF34)
42	COUPLING
43	WATER ORIFICE
44	BOLT
45	PLATE G1
46	HEAT SHIELD G2
47	HEAT SHIELD G3
48	HEAT SHIELD G4
49	HEAT SHIELD G5
50	HEAT SHIELD G6
51	HEAT SHIELD G7
52	HEAT SHIELD G8
53	HEAT SHIELD G9
54	SPLASH SHIELD G10
55	SPLASH SHIELD G11
56	LOCATING FRAME
57	SHEAR SPACER
58	SUPERHEAT PIPE SUPPORT P1
59	SUPERHEAT PIPE SUPPORT P2
60	PLATE P1
61	PLATE P2
62	PLATE P3
63	BARRIER P1
64	BARRIER P2
65	SPACER
66	STEAM RISER CLAMP STOP P1
67	STEAM RISER CLAMP STOP P2
68	STEAM RISER CLAMP P1
69	STEAM RISER CLAMP P2
70	HEX HD CAP SCREW
71	HEX HD CAP SCREW
72	HEX HD CAP SCREW
73	HEX HD CAP SCREW
74	NUT
75	NUT
76	STEAM DIFF COUP ROTATION STOP
77	6 IN. INST GUIDE TUBE ROTATION STOP P1
78	6 IN. INST GUIDE TUBE ROTATION STOP P2
79	SHOULDER BOLT
80	WASHER
81	HEX HD CAP SCREW

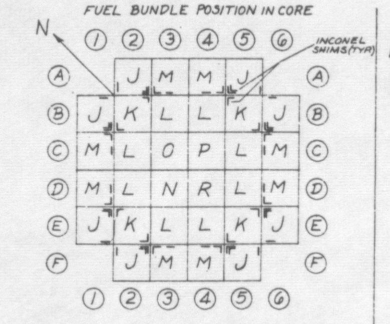
PART NO.	NAME
82	HEX HD CAP SCREW
83	HEX HD CAP SCREW
84	HOLD DOWN PLATE
85	CAPSULE & MOUNT G1
86	CAPSULE & MOUNT G2
87	CAPSULE & MOUNT G3
88	CAPSULE & MOUNT G4
89	MONITOR INSERTION GUIDE
90	CORNER SPACER
91	WASHER
92	HEX HD CAP SCREW
93	AUXILIARY SCRAM ROD DRIVE
94	BLIND OUTLET CAP
95	CONTROL ROD
96	SCREW
97	NEUTRON SOURCE
98	LATERAL SUPT.
99	LATERAL SUPT. GUIDE

922-26

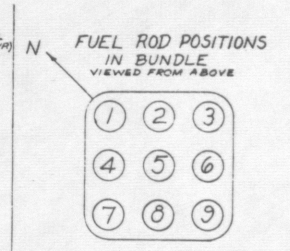
FIGURE 4.1
REACTOR ASSEMBLY ELEVATION

	FUEL ROD DESIGNATION									
	ROD #1 (CORNER)	ROD #2 (RADIAL)	ROD #3 (CORNER)	ROD #4 (RADIAL)	ROD #5 (CENTER)	ROD #6 (RADIAL)	ROD #7 (CORNER)	ROD #8 (RADIAL)	ROD #9 (CORNER)	FUEL BUNDLE DESIGNATION
A2	J-1-7.5	J-2-7.5	J-3-4.0	J-4-5.4	J-5-7.5	J-6-5.4	J-7-5.4	J-8-4.0	J-9-4.0	A2-J- -I(.093)
A3	M-1-5.4	M-2-5.4	M-3-5.4	M-4-5.4	M-5-7.5	M-6-7.5	M-7-4.0	M-8-4.0	M-9-4.0	A3-M- -I(.093)
A4	M-1-5.4	M-2-5.4	M-3-5.4	M-4-7.5	M-5-7.5	M-6-5.4	M-7-4.0	M-8-4.0	M-9-4.0	A4-M- -I(.093)
A5	J-1-4.0	J-2-7.5	J-3-7.5	J-4-5.4	J-5-7.5	J-6-5.4	J-7-4.0	J-8-4.0	J-9-5.4	A5-J- -I(.093)
B1	J-1-7.5	J-2-5.4	J-3-5.4	J-4-7.5	J-5-7.5	J-6-4.0	J-7-4.0	J-8-5.4	J-9-4.0	B1-J- -I(.093)
B2	K-1-7.5	K-2-7.5	K-3-4.0	K-4-7.5	K-5-7.5	K-6-4.0	K-7-4.0	K-8-4.0	K-9-4.0	B2-K- -I(.050)
B3	L-1-4.0	L-2-5.4	L-3-5.4	L-4-4.0	L-5-5.4	L-6-7.5	L-7-4.0	L-8-5.4	L-9-5.4	B3-L-Z-
B4	L-1-5.4	L-2-5.4	L-3-4.0	L-4-7.5	L-5-5.4	L-6-4.0	L-7-5.4	L-8-5.4	L-9-4.0	B4-L-Z-
B5	K-1-4.0	K-2-7.5	K-3-7.5	K-4-4.0	K-5-7.5	K-6-7.5	K-7-4.0	K-8-4.0	K-9-4.0	B5-K- -I(.050)
B6	J-1-5.4	J-2-5.4	J-3-7.5	J-4-4.0	J-5-7.5	J-6-7.5	J-7-4.0	J-8-5.4	J-9-4.0	B6-J- -I(.093)
C1	M-1-5.4	M-2-5.4	M-3-4.0	M-4-5.4	M-5-7.5	M-6-4.0	M-7-5.4	M-8-7.5	M-9-4.0	C1-M- -I(.093)
C2	L-1-4.0	L-2-4.0	L-3-4.0	L-4-5.4	L-5-5.4	L-6-5.4	L-7-5.4	L-8-7.5	L-9-5.4	C2-L-Z-
C3	O-1-4.0	O-2-5.4	O-3-5.4	O-4-5.4	O-5-5.4	O-6-5.4	O-7-5.4	O-8-5.4	O-9-4.0	C3-O-Z-
C4	P-1-5.4	P-2-5.4	P-3-4.0	P-4-5.4	P-5-5.4	P-6-5.4	P-7-4.0	P-8-5.4	P-9-5.4	C4-P-Z-
C5	L-1-4.0	L-2-4.0	L-3-4.0	L-4-5.4	L-5-5.4	L-6-5.4	L-7-5.4	L-8-7.5	L-9-5.4	C5-L-Z-
C6	M-1-4.0	M-2-5.4	M-3-5.4	M-4-4.0	M-5-7.5	M-6-5.4	M-7-4.0	M-8-7.5	M-9-5.4	C6-M- -I(.093)
D1	M-1-5.4	M-2-7.5	M-3-4.0	M-4-5.4	M-5-7.5	M-6-4.0	M-7-5.4	M-8-5.4	M-9-4.0	D1-M- -I(.093)
D2	L-1-5.4	L-2-7.5	L-3-5.4	L-4-5.4	L-5-5.4	L-6-5.4	L-7-4.0	L-8-4.0	L-9-4.0	D2-L-Z-
D3	N-1-5.4	N-2-5.4	N-3-4.0	N-4-5.4	N-5-5.4	N-6-5.4	N-7-4.0	N-8-5.4	N-9-5.4	D3-N-Z-
D4	R-1-4.0	R-2-5.4	R-3-5.4	R-4-5.4	R-5-5.4	R-6-5.4	R-7-5.4	R-8-5.4	R-9-4.0	D4-R-Z- -FW-T/C 3,4,5,6-V
D5	L-1-5.4	L-2-7.5	L-3-5.4	L-4-5.4	L-5-5.4	L-6-5.4	L-7-4.0	L-8-4.0	L-9-4.0	D5-L-Z- -FW-T/C 3,4,5,6
D6	M-1-4.0	M-2-7.5	M-3-5.4	M-4-4.0	M-5-7.5	M-6-5.4	M-7-4.0	M-8-5.4	M-9-5.4	D6-M- -FW-T/C 3,4,5,6-I(.093)
E1	J-1-4.0	J-2-5.4	J-3-4.0	J-4-7.5	J-5-7.5	J-6-4.0	J-7-7.5	J-8-5.4	J-9-5.4	E1-J- -I(.093)
E2	K-1-4.0	K-2-4.0	K-3-4.0	K-4-7.5	K-5-7.5	K-6-4.0	K-7-7.5	K-8-7.5	K-9-4.0	E2-K- -I(.050)
E3	L-1-4.0	L-2-5.4	L-3-5.4	L-4-4.0	L-5-5.4	L-6-7.5	L-7-4.0	L-8-5.4	L-9-5.4	E3-L-Z-
E4	L-1-5.4	L-2-5.4	L-3-4.0	L-4-7.5	L-5-5.4	L-6-4.0	L-7-5.4	L-8-5.4	L-9-4.0	E4-L-Z- -T/C 3,5,7
E5	K-1-4.0	K-2-4.0	K-3-4.0	K-4-4.0	K-5-7.5	K-6-7.5	K-7-4.0	K-8-7.5	K-9-7.5	E5-K- -FW-T/C 3,5,7-I(.050)
E6	J-1-4.0	J-2-5.4	J-3-4.0	J-4-4.0	J-5-7.5	J-6-7.5	J-7-5.4	J-8-5.4	J-9-7.5	E6-J- -FW-T/C 3,5,7-I(.093)
F2	J-1-5.4	J-2-4.0	J-3-4.0	J-4-5.4	J-5-7.5	J-6-5.4	J-7-7.5	J-8-7.5	J-9-4.0	F2-J- -I(.093)
F3	M-1-4.0	M-2-4.0	M-3-4.0	M-4-5.4	M-5-7.5	M-6-7.5	M-7-5.4	M-8-5.4	M-9-5.4	F3-M- -I(.093)
F4	M-1-4.0	M-2-4.0	M-3-4.0	M-4-7.5	M-5-7.5	M-6-5.4	M-7-5.4	M-8-5.4	M-9-5.4	F4-M- -T/C 3,5-I(.093)
F5	J-1-4.0	J-2-4.0	J-3-5.4	J-4-5.4	J-5-7.5	J-6-5.4	J-7-4.0	J-8-7.5	J-9-7.5	F5-J- -T/C 3,5-I(.093)
5 SPARE BUNDLES LISTED BELOW										
1 BUNDLE REQUIRED	L-1-5.4 T/C 1 & 2	L-2-7.5	L-3-5.4	L-4-5.4 T/C 1 & 2	L-5-5.4 T/C 1 & 2	L-6-5.4	L-7-4.0	L-8-4.0	L-9-4.0	D5-L- -FW-T/C 1,2,3,4,5,6,7 POTTER FLOWMETERS(3)
2 BUNDLES REQUIRED	M-1-5.4	M-2-5.4	M-3-4.0	M-4-5.4	M-5-7.5	M-6-4.0	M-7-5.4	M-8-7.5	M-9-4.0	C1-M- -I(.093)
2 BUNDLES REQUIRED	L-1-4.0	L-2-4.0	L-3-4.0	L-4-5.4	L-5-5.4	L-6-5.4	L-7-5.4	L-8-7.5	L-9-5.4	C2-L-Z-

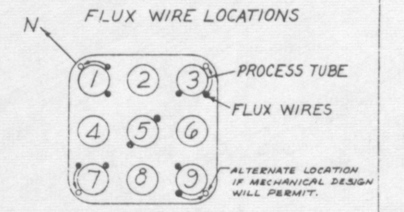
FUEL BUNDLE POSITION IN CORE



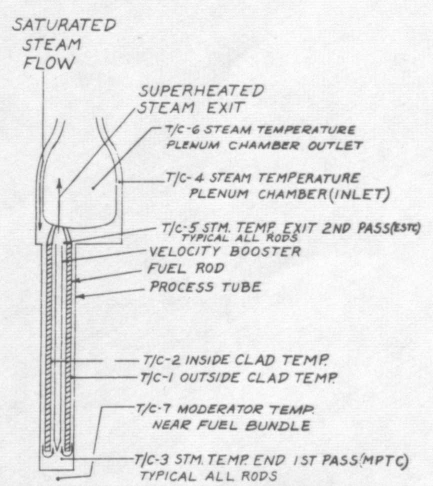
FUEL ROD POSITIONS IN BUNDLE VIEWED FROM ABOVE



FLUX WIRE LOCATIONS



THERMOCOUPLE LOCATIONS



CLAD MATERIAL CODE

J -304L STN STL DOUBLE VACUUM MELTED
K -310 STN STL DVM
L -INCOLOY
M -INCONEL
N -304 COM'L STN STL
O -316 STN STL
P -347 STN STL
R -348 STN STL

FUEL BUNDLE DESIGNATION (EXAMPLE)

A2-J-Z-1.000-FW-TC-V-I(050)

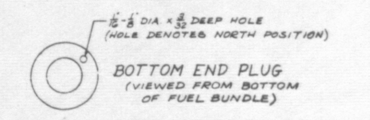
- BUNDLE POSITION IN CORE
- CLAD MATERIAL
- METALLOGRAPHIC SAMPLES INSTALLED INSIDE BOOSTER TUBE OF ROD #5
- ORIFICE SIZE (INCHES)
- FLUX WIRES
- THERMOCOUPLES
- VIBROMETER
- INCONEL SHIMS-THICKNESS IN INCHES

FUEL ROD DESIGNATION

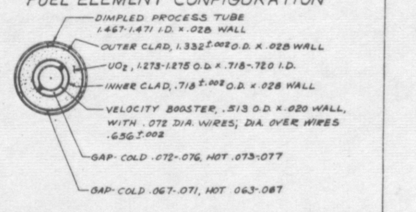
J-1-3.5-TC (EXAMPLE)

- FUEL ROD T/C'S #1 & 2
- % UO_2 ENRICHMENT
- FUEL ROD POSITION IN BUNDLE
- FUEL ROD CLAD MATERIAL

ORIENTATION OF FUEL ROD IN BUNDLE

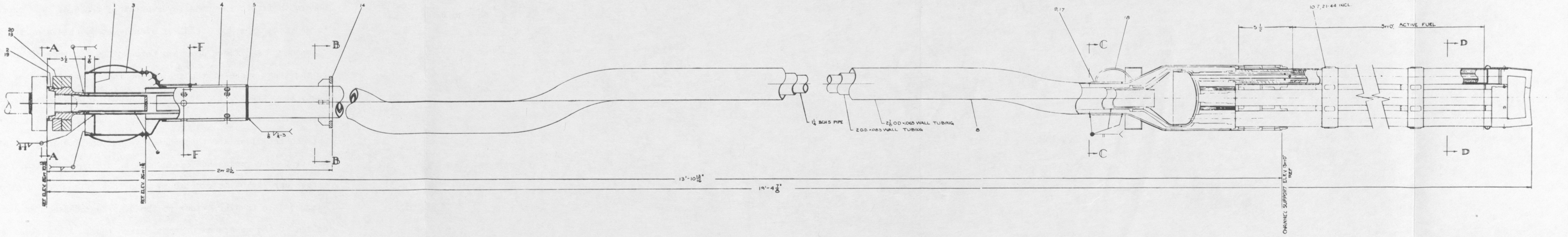


FUEL ELEMENT CONFIGURATION

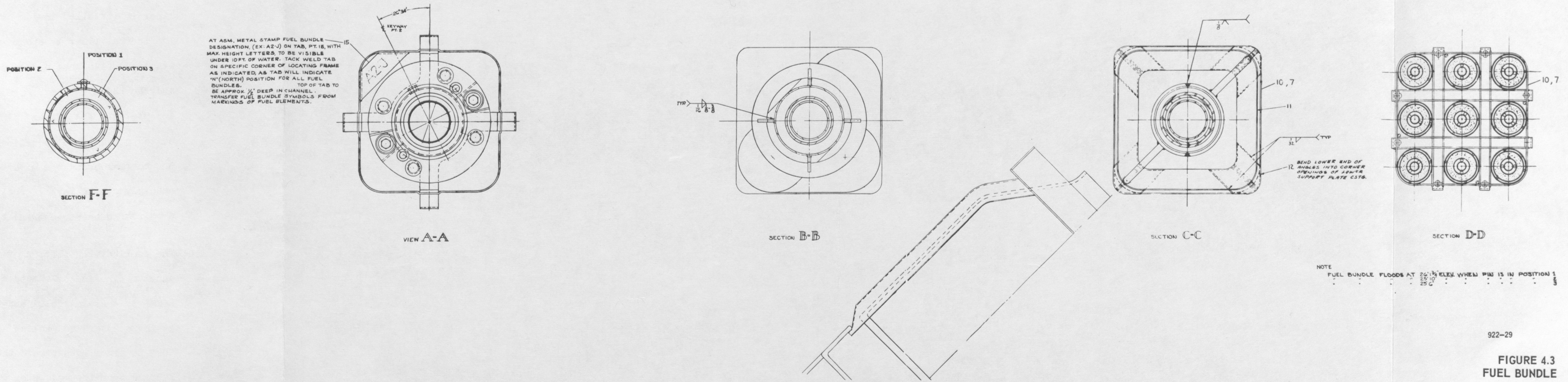


922-36

FIGURE 4.2
MASTER FUEL INDEX



PART NO.	NAME
1	CPLG. SPPT.
2	FUEL FLANGE
3	LOCATING FRAME
4	ORIFICE SLEEVE
5	RETAINER
6	ORIFICE SLEEVE
7	FUEL ELEMENT
8	RISER AND DOWNCOMER
9	WELD SLEEVE
10	FUEL ELEMENT
11	COLLECTOR
12	ANGLE
13	CLAMP
14	COLLAR
15	INDENT. TAB
16	RISER AND DOWNCOMER
17	HALF SLEEVE
18	HANDLE
19	FUEL FLANGE
20	CLAMP
21-44	FUEL ELEMENT



922-29

FIGURE 4.3
FUEL BUNDLE

The "riser-downcomer" assembly is composed of three concentric tubes separated from one another by mechanical spacers. The outer tube is formed from Type 304 stainless steel. Steam enters at the top of the "downcomer" and flows downward between the "downcomer" and "riser" tubes to the steam space at the top of the fuel element cluster. The center tube, the "riser", is formed from Inconel because the use of this material, rather than stainless steel, minimizes the thermal expansion mismatch between the riser and the reactor vessel and its internal piping. In addition, the resistance of Inconel to ordinary chemical corrosion and stress cor rosion is excellent. The lower flange of the Marman "Conoseal" used to join the fuel riser to the superheated steam jumper pipe is welded to the upper end of the "riser" tube. A stainless steel thermal liner is installed inside of, and concentric with, the "riser" pipe. The thermal liner is spaced away from the "riser" by a stainless wire wrap spacer so that a dead steam space is created between the "riser" and the thermal liner. The dead steam space acts as a thermal insulator and prevents the "riser" pipe from reaching the temperature of the superheated steam. The reduction in "riser" temperature assists in reducing the thermal expansion mismatch discussed above. The "riser-downcomer" assembly of the Mark II fuel bundle is attached to the fuel cluster by a pair of concentric transition pieces and tube sheets.

The fuel bearing portion of the Mark II fuel bundle consists of nine annular-type fuel elements arranged in a 3 by 3 array. Because the EVESR is a steam-cooled and water-moderated type of nuclear reactor, it is necessary to enclose each fuel element inside of a metal process tube in order to separate the cooling steam surrounding the fuel element surface from the moderator water. The process tube type of fuel bundle may be designed so as to provide either two-pass series-flow, or single-pass parallel-flow coolant passages by the manner of arrangement of the process tubes and the tube sheets in a specific bundle. The fuel bundles for the first core load for the EVESR are all of the two-pass series-flow type. The saturated steam is distributed to the nine fuel elements by means of a steam plenum formed between the upper and lower tube sheets. The lower tube sheet supports the process tubes and is attached to a frame that supports and positions the bundle on the fuel channel surrounding the fuel bundle. The process tube is formed from Type 304 stainless steel of 0.028-inch thickness and is welded closed at the lower end and welded to the tube sheet at the top. The process tube is spaced from the fuel rod by dimples formed on the process tube itself. A Type 304 stainless steel nose piece supports the lower end of the process tubes and guides and protects the process tubes during refueling. The upper end of the fuel element is joined to the upper tube sheet by an adapter tube welded to the fuel element top end plug and to the tube sheet. The arrangement of tube sheets and the annular configuration of the fuel element and process tube form a steam flow path that is first downward over the outside surface of the fuel element and then upward over the inside surface. A stainless steel velocity booster tube is centered in the inside of the fuel element to reduce the second pass flow area and increase the velocity of the cooling steam at that point. The velocity booster is spaced away from the inner surface of the fuel element by two wires that are wrapped spirally around the velocity booster tube and spot-welded along the length of the wire.

Each Mark II fuel element is formed from two concentric tubes containing a stack of hollow, sintered UO_2 pellets of relatively low enrichment. The concentric cladding tubes are 0.028-inch in thickness regardless of the cladding material used in any bundle. The cladding tubes are sealed at both ends by suitable welded end plugs. The annular UO_2 pellets are ground internally and externally and have a 0.005 to 0.010-inch deep circumferential depression formed at one end in order to minimize the possibility of excess UO_2 longitudinal expansion relative to the clad expansion. A fission gas chamber is provided in the upper end plug of the fuel element. The fuel rod is designed for a 20 percent fission gas release. The external clad material is swaged down over the ground pellet to give a maximum cold diametrical gap between the pellet and the cladding of 0.002 inch. The inside clad material is expanded against the inside surface of the UO_2 pellet so that a maximum cold gap of 0.002 inch exists. This inner cladding expansion operation is required in order to improve the heat transfer in the second pass of the fuel element.

Power flattening (rod-to-rod power variations) within each individual fuel bundle is accomplished by the use of three different UO_2 enrichments and by the use of Inconel strips attached to the fuel bundle. The locations of these Inconel strips are shown in the "Fuel Bundle Position In Core" section of Figure 4.2.

Metallographic samples (about 15 to 20 per velocity booster tube) of different superheat clad materials are located inside the central velocity booster tube of 12 of the 32 fuel bundles. These samples, taken from sections of fuel rod cladding, are approximately 2 inches long by one-half inch wide (maximum dimension) by 1/16 inch thick, and are used to determine the effects of irradiation at elevated temperatures on the physical properties of the materials. These samples are fastened to a thin sheet metal strap suspended inside the booster tube.

Flux wires installed inside 1/8-inch-diameter tubes that are fastened to the fuel bundles and process tubes are used for experimentally determining the relative power distribution within a typical fuel bundle. The locations of the flux wires are shown on Figure 4.2. Thermocouples for measuring steam or moderator temperatures are located in selected positions within the various fuel bundles as shown in Figure 4.2. The thermocouples are bundled together and routed vertically from the point of attachment to the fuel bundle, along the riser-downcomer tubes, to a position below the jumper flange where they then travel horizontally and exit from the reactor vessel through one of the vessel instrument flanges.

A number of mechanical tests have been performed using prototype fuel bundle components to demonstrate the reliability of the mechanical design of the Mark II fuel bundle under simulated reactor operating and abnormal conditions.

Collapse tests, at the operating temperature of 650°F, on a dimpled process tube fuel element assembly indicated that the Mark II process tube will not collapse up to a differential pressure across the tube of 230 psi. Only 21 percent of the outer annular volume collapsed at this pressure. At a differential pressure of 550 psi, 65 percent of the outer annular volume collapsed. Similar tests conducted on an assembly spaced with spiral wires, a design which may be used in later fuel designs, collapsed at 390 psi with 31 percent of the outer volume collapsing. In no case would steam flow have been completely closed off. The normal pressure differential existing across the process tube is less than 10 psi. The pressure differential across the process tube under the extreme condition of a pipeline break outside the reactor vessel is 137 psi.

It is recognized that the Mark II fuel elements will tend to bow within each fuel channel due to the thermal gradient across the diameter of each element. Bowing tests, together with analysis, were performed to determine local heat transfer effects on cladding temperature, strain effects on the cladding, and whether any intereference with control rod functioning is possible as a result of the structural distortion. One test was performed with a three-foot-section of annular fuel (without process tube) to correlate temperature gradient across the annular fuel element with bow deflection of the element caused by the temperature gradient. The predicted deflection and the experimentally measured deflection, as well as the predicted and experimentally measured value of the force required to restore the deflection to zero, agreed very closely. As expected, it was found that the fuel pellets had little influence on the bowing of the fuel element.

A full length fuel assembly test was performed to investigate the likelihood that a fuel element in its process tube could bow sufficiently to bulge the sides of its fuel channel and thus interfere with control rod operation. The fuel assembly included simulated process tube spacers and the restraint due to the upper tube sheet welds. The fuel element was loaded by exerting force on the fuel element through drilled openings in the process tubes. The forces exerted were consistent with the force that had been determined to be necessary to restrain bowing in the most severely bowed Mark II fuel elements. The deflection of the fuel assembly was insufficient to cause the process tube to touch the side of a fuel channel, and it therefore was concluded that a bowing fuel element cannot interfere with control rod operation.

Additional fuel element tests were run to determine the degree to which fuel element bowing could close the fuel element to process tube steam gap and by so doing interfere with cooling of the fuel cladding. It was determined that the local forces expected in the Mark II core due to thermal distortions could not deform the fuel element to process tube steam gap as much as 10 percent.

Thermal shock tests of the fuel element end closures were conducted by heating the end closures up to $800^{\circ}F$ and then immersing the heated part in $100^{\circ}F$ water. No measurable distortions or fissures were found anywhere on the element following 5 quenchings of the type described. The thermal shock test is an important demonstration of the reliability of the fuel element to withstand the temperature transient that the fuel element undergoes if it should be flooded while at operating temperature. It is expected that the actual temperature difference between the water and the fuel element will be less than $70^{\circ}F$ at the time flooding normally occurs in preparation for refueling. A number of fuel development tests of usefulness in the design of the Mark II fuel have been conducted as part of the AEC Nuclear Superheat Project by General Electric under Project

Agreement 13 of Contract Number AT(04-3)-189. These development tests include basic tests and analyses in the mechanisms of wrinkle formation of thin wall tubing, plastic strain cycling of wrinkles both in and out-of-pile, studies of plastic strain interference between the UO_2 and outer cladding, and a series of corrosion and metallurgical type of tests. Various progress and topical reports covering this work can be found in the Eleventh Quarterly Progress Report of the Nuclear Superheat Project (GEAP-3924).

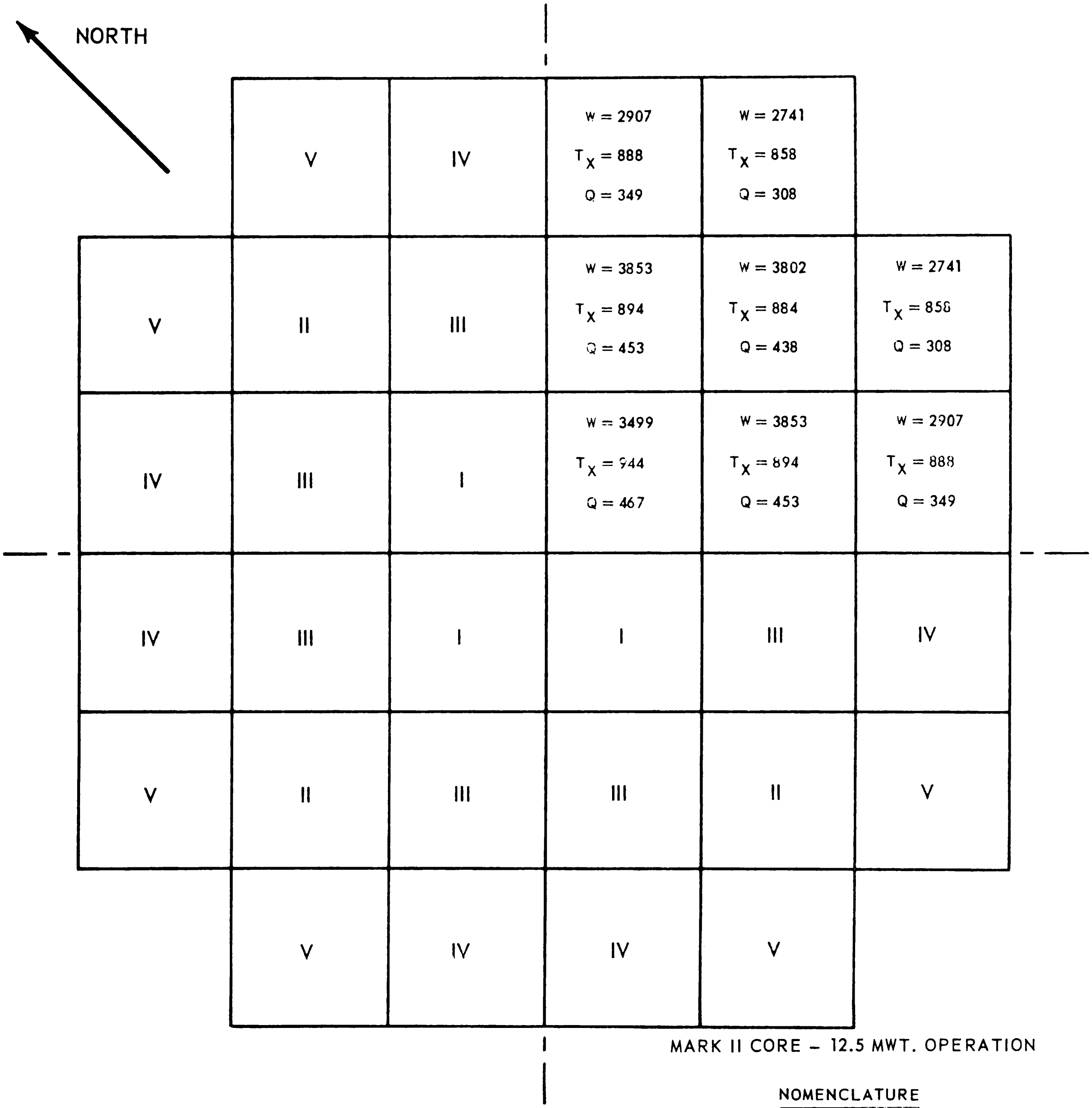
An out-of-pile loop facility for flow testing full-scale EVESR fuel bundles was constructed by General Electric at the Pacific Gas & Electric Company's Moss Landing Steam Plant. Steam at up to 1350 psig and 950°F is available.

4.3 Heat Transfer and Fluid Flow

4.3.1 General Characteristics

This section describes the calculated thermal characteristics of the Mark II core. For the purpose of thermal-hydraulic analysis, the core consists of two regions: the superheat steam circuit and the moderator circuit. The core is composed of 32 fuel bundles and these, with respect to thermal-hydraulic properties, are divided into five zones (I through V) shown on Figure 4.4.

The steam flow in each bundle is adjusted, with respect to the thermal zone in which it is located, to give the desired surface temperature in the hottest fuel element within the bundle. In practice, steam flow adjustment is provided through the use of a combination of remote-operated valves outside the reactor vessel and semi-permanent orifices



MARK II CORE – 12.5 MWT. OPERATION

NOMENCLATURE

W – BUNDLE FLOW - LB/HR

T_X – BUNDLE MIXED MEAN EXIT STEAM TEMP, DEG F

Q – BUNDLE POWER - KW

FIGURE 4.4

THERMAL ZONE OUTLET CONDITIONS

922–22

located in the steam risers. The outlet temperature from each zone will be monitored. The flow rates in all zones will also be measured as shown on Figure 4.5. Note on Figure 4.4 that the steam outlet temperature can be different for each zone. It is planned to measure the power distribution and then correlate this to the steam temperatures obtained from an instrumented fuel bundle as a confirmation of the analytical techniques being used.

Table 4.1 summarizes the basic reactor parameters for operation of the Mark II core at a thermal power level of 12.5 MW. Also included in Table 4.1 are the important parameters of the hottest element in the hottest of the five thermal zones which is element number 10, Zone II. The operating condition for all 39 of the unique fuel element locations, including element number 10, are shown on Figure 4.6. Since 1/8-core symmetry is maintained, Figure 4.6 is representative of the entire Mark II core.

The selection of 1250°F peak clad temperature is somewhat arbitrary and is not the only factor influencing the thermal power limit for the fuel. The thermal power limit is influenced by complex fuel and cladding dimensional relationships which are also a function of specific power. The criteria for establishing an upper limit are not well defined. Initial operation of EVESR will take place in stepwise increases up to 1250°F clad temperature. Since one of the purposes of the reactor is to investigate such things as realistic temperature and specific power limitations, it is anticipated also that future further stepwise increases

Table 4.1

Reactor Data Sheet

Overall Reactor Data		
Reactor thermal power	Mwt	12.5
Power added to steam (net superheat)	Mw	7.41
Power added to moderator (heat transferred through process tube)	Mw	4.47
Power added to moderator (nuclear)	Mw	0.62
*Steam flow from boiler	lb/hr	90,600
Steam flow from moderator	lb/hr	14,600
Steam flow through superheat passages	lb/hr	105,200
Steam inlet pressure (moderator pressure)	psig	960
Reactor steam outlet temperature (main header)	^{o}F	870
Reactor steam ΔP (nozzle to nozzle)	psi	20
Inlet steam temperature	^{o}F	543
Outlet steam temperature (core mixed mean)	^{o}F	890
Make-up water inlet temperature	^{o}F	130
Moderator average exit void fraction	% by volume	11.8
Moderator average core void fraction	% by volume	7.1
Moderator recirculation rate	lb/hr	2,144,000
Reactor Core		
Number of fuel assemblies (bundles)		32
Number of elements per assembly		9
Total number of fuel elements		288
Number of control blades		12
Number of auxiliary scram blades		4
Control rod pitch	inches	9.334
Circumscribed core diameter	inches	58.2
Equivalent core diameter	inches	42.1
Active core length	inches	60
Core length to equivalent diameter ratio		1.42
Fuel volume	ft^3	9.02
Core volume	ft^3	48.4
Total weight of UO_2 in core	kg	2523
Average UO_2 pellet density (% of theoretical)	%	93.5
Fuel	Sintered UO_2	
Fuel geometry	Annular pellets-process tube	
Process tube O.D.	inches	1.525
Process tube wall thickness	inch	0.028
Outer clad O.D.	inches	1.332
Outer clad thickness	inch	0.028
Inner clad O.D.	inch	0.718

* Fossil fired boiler rated at 120,000 lb/hr.

Table 4.1 (Con'd)

Reactor Core (con'd)		
Inner clad thickness	inch	0.028
Fuel pellet O.D.	inches	1.275
Fuel pellet I.D.	inch	0.719
Velocity booster O.D.	inch	0.513
Velocity booster wall thickness	inch	0.020
Outer flow annulus gap	inch	0.065
Inner flow annulus gap	inch	0.075
Core Heat Transfer***		
Heat transferred from fuel (thermal)	Btu/hr	40,534,000
Total heat transfer area	ft^2	757
Average core heat flux	Btu/ft^2hr	53,600
*Maximum core heat flux	Btu/ft^2hr	110,200
*Maximum to average core heat flux		2.1
Average specific power in fuel	kw/kg	4.7
*Maximum specific power in fuel	kw/kg	8.5
*Maximum to average specific power		1.83
Average Core Power density	kw/liter	8.7
**Allowable clad temperature (steady state design)	^oF	1250
Hot element most probable peak clad temp.	^oF	1100
**Hot element maximum clad temp.	^oF	1250
**Hot element maximum fuel temp.	^oF	<2000
Hot element first pass velocity (avg.)	fps	32
Hot element second pass velocity (avg.)	fps	86
Hot element first pass steam inlet temp.	^oF	540
Hot element second pass steam inlet temp.	^oF	735
Hot element out let steam temp.	^oF	946
Hot element first pass pressure drop	psi	3.1
Hot element second pass pressure drop	psi	11.2

* Does not include statistical uncertainties in heat generation.

** Includes statistical uncertainties--see discussion on cladding temperatures in this section.

*** The data presented on core performance corresponds to the power distribution shown on Figures 4.4 and 4.6. For detailed data within each bundle, see Figure 4.8.

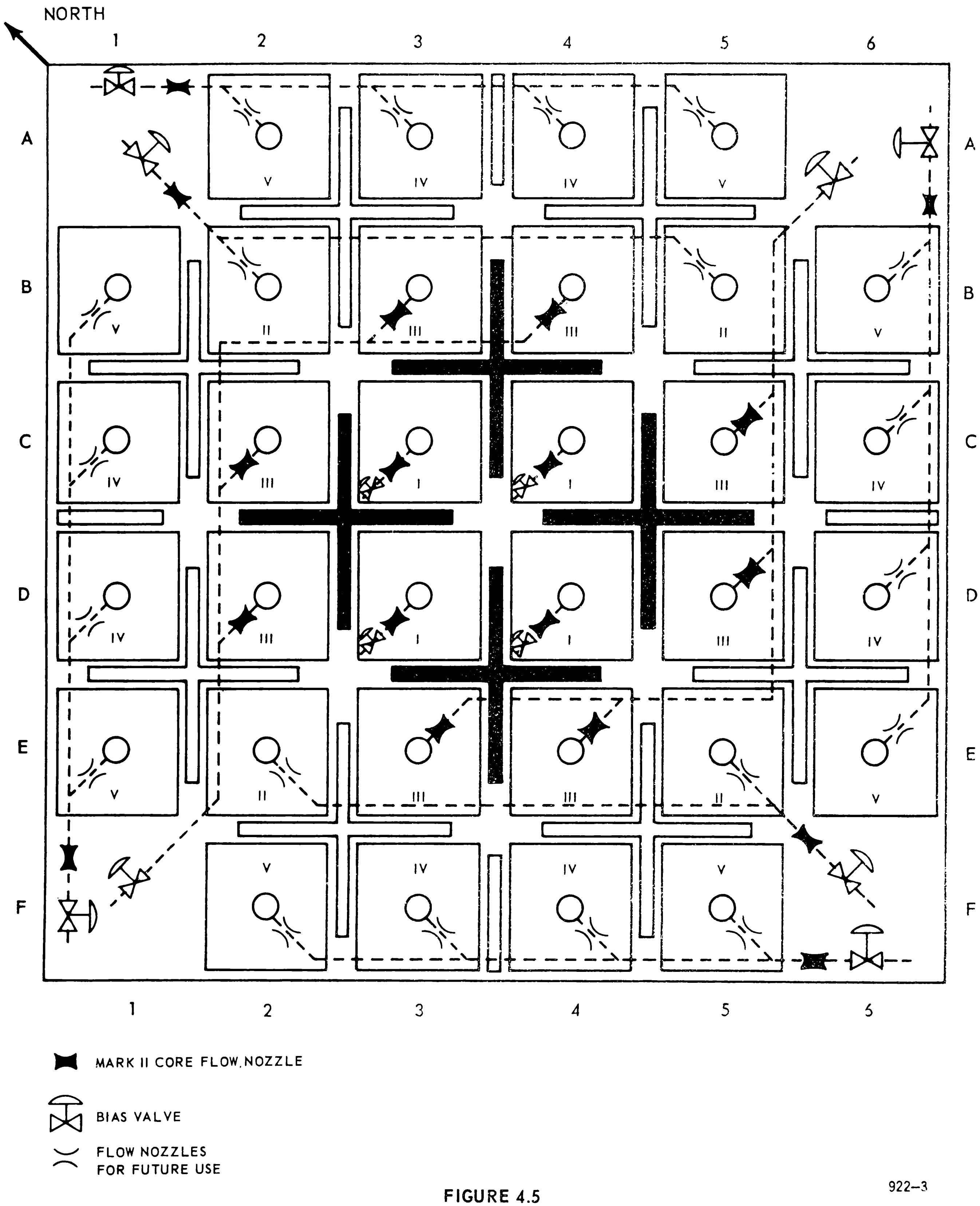


FIGURE 4.5
BIAS VALVE AND FLOW NOZZLE ARRANGEMENT

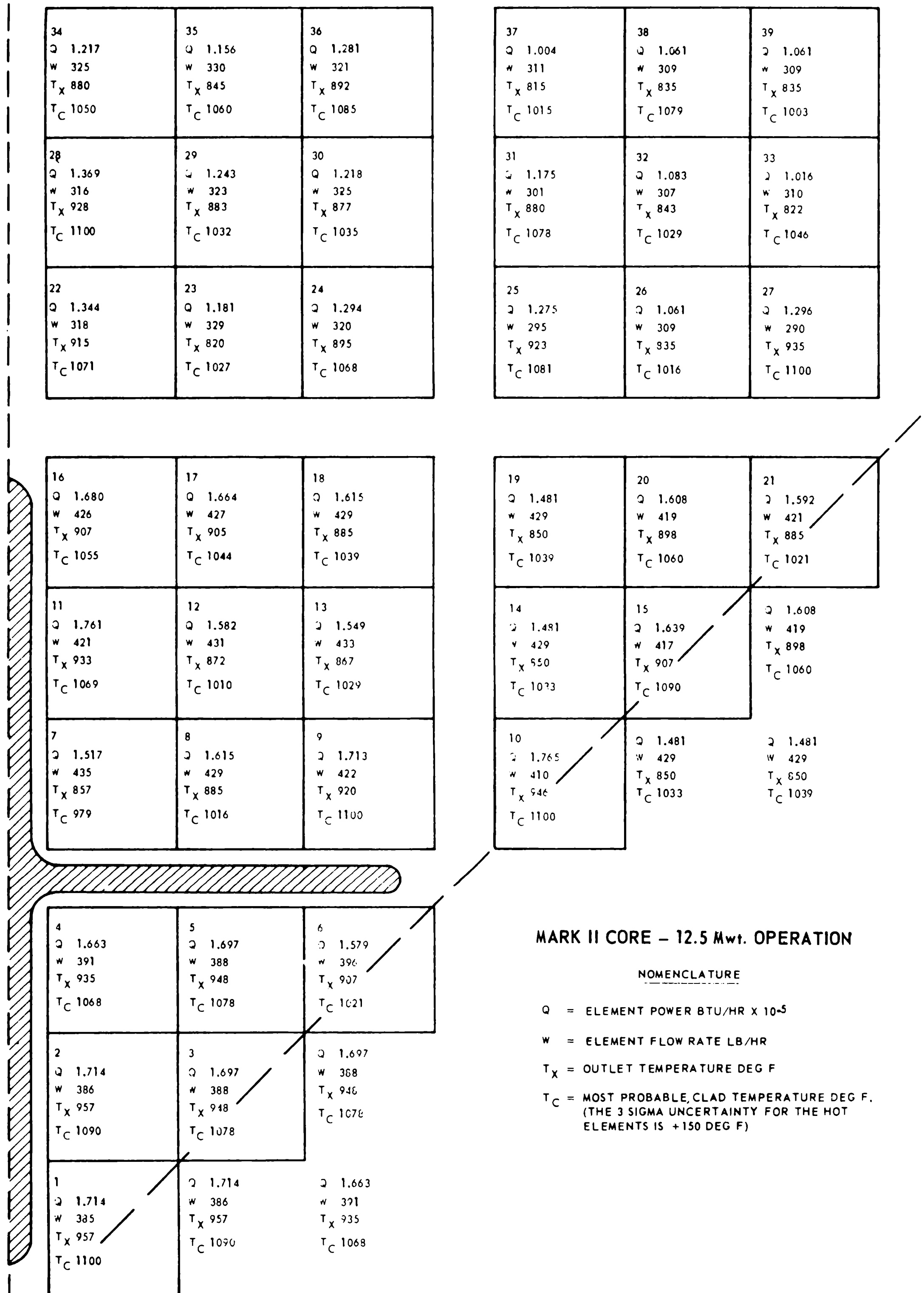


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FIGURE 4.6
FUEL ELEMENT OPERATING CONDITIONS

will be made to cladding temperature above $1250^{\circ}F$ on a planned experimental basis and dependent on previous results.

4.3.2 Superheated Steam Region

Each of the fuel elements in the core was analyzed by means of a two-dimensional computer code which considers: variation of axial heat generation; radial flux depression through the fuel meat; local heat transfer coefficients, based on the local properties of the surface of interest; clad-to-fuel contact and clad thermal impendances; thermal radiation to the process tube; the process tube end and its outer film thermal impedances; and pressure drop due to friction, discontinuities, entrance and exit effects, as well as compressibility effects, all based on local steam properties.

Axial heat transfer in either the fuel or clad was not included because hand calculations indicated that only a slight error was introduced by neglecting it. Axial variation of material physical properties are not included except for the steam, since these variations can have only second order effects. The model is two dimensional and circumferential effects are included by hand as explained later in this section. Uniform fuel thermal conductivity is assumed, but this assumption is not important since fuel center melting is not limiting for full power operation of the EVESR.

The heat transfer coefficient is calculated from the following correlation:

$$N_{Nu} = 0.0197\,(N_{Re})^{0.8}\,(N_{Pr})^{1/3}\left(\frac{T_b}{T_w}\right)^{0.5}\left(1 + \frac{K}{L/D}\right) \text{ for } N_{Re} > 2000$$

$$N_{Nu} = 0.26\,(N_{Re})^{1/3}\,(N_{Pr})^{1/3} \qquad \text{for } N_{Re} < 2000$$

Where: N_{Nu} is the local Nusselt number

N_{Re} is the local Reynolds number

N_{Pr} is the local Prandtl number

T_w is the local wall surface temperature, ^oF

T_b is the local bulk steam temperature, ^oF

L is the distance from the entrance to the point of interest
D is the hydraulic diameter
K is a constant depending on the entry length
K is 0.6 for the first pass
and 1.2 for the second pass

Both the Reynolds and Prandtl Numbers are evaluated at the bulk temperature. The equation includes a correction(1) to account for the effect of heating on only one side of the annulus. This gives results that are consistent with recent tests on superheated steam in narrow annuli(2), rectangular channels, and tubes(3). Reserve flow is available from the boiler should it be required for unforeseen heat transfer reasons.

1. "Heat Transfer With Turbulent Flow in Concentric and Eccentric Annuli With Constant and Variable Heat Flux", E. Y. Leung, W. M. Kays, W. C. Reynolds, Report AHT-4, Standford University, Standford, California, April 15, 1962.

2. "Heat Transfer to Superheated Steam in a Thin Annulus", K. F. Newen, G. J. Kangas, N. C. Sher, American Institute of Chemical Engineers Heat Transfer Conference, August 5-8, 1962. Paper #23.

3. "An Experimental Investigation of Heat Transfer to Superheated Steam in Round and Rectangular Channels", J. B. Heinman, Report ANL-6213, September, 1960.

The pressure drop for each steam pass was calculated on a node-to-node basis from the equation:

$$\Delta P = \frac{4fl}{D}\left(\frac{W}{A}\right)^2 \frac{v_a}{2g(144)} + \left(\frac{W}{A}\right)^2 \frac{(v_o - v_i)}{(g)(144)} + K\left(\frac{W}{A}\right)^2 \frac{v}{2g(144)}$$

Where: $f = \frac{0.079}{(N_{Re})^{.25}}$ for N_{Re} 2000

$f = \frac{16}{N_{Re}}$ for N_{Re} 2000

W = Fuel element flow rate, lb/sec

L = Fuel element region length, ft

D = Hydraulic diameter, ft

A = Flow area, ft^2

v = Specific volume at any point, ft^3/lb

K = Local loss coefficient

Subscript "o" for outlet

Subscript "i" for inlet

Subscript "a" for average

g = gravity acceleration, 32.2 ft/sec^2

The validity of these equations for both smooth tubes and annuli has been demonstrated with air and other gases. To allow for the effect of the process tube dimples on the first pass, the friction factor was increased by 25 percent. It should be noted that the absolute value of pressure drop across the core is relatively unimportant as long as the relative pressure drops between elements is calculated on a consistent basis.

The emissivity between the outer clad and process tube has been measured by several experimenters (1)(2). A constant value of 0.5 was used in all calculations.

The contact coefficient between the inside clad and fuel surface of 300 Btu/ft^2hr^oF was determined by calculations. This value is based on the statistical manufacturing tolerances expected to be achieved in the fabrication of the fuel elements and was correlated with the contact coefficients that were calculated for the superheat fuel elements irradiated in the SADE* loop. The SADE loop fuel elements were instrumented so that the first and second pass steam temperature rise could be measured and separated. The contact coefficient for the SADE element was calculated from these experimental data. The contact coefficient of the outer clad to fuel surface was taken as 1000 Btu/ft^2hr^oF.

The radial flux depression through the fuel was calculated to be about 1.25. This factor does not have a strong influence on the analysis. The axial power shape of the core is shown on Figure 4.7.

The thermal conductivity of the UO_2 was based on data published in "Nucleonics", June 1961, Vol. 19, page 83. An average value of 1.6 Btu/ft^2hr^oF was selected for the EVESR calculations. This value covers

1. GEAP-3319, "Superheat Process Tube Heat Transfer Tests" by E. Janssen, January 5, 1962.

2. DMIC Memorandum 111 (OTS PB 171630), "The Emittance of Stainless Steels" by W. D. Wood, H. W. Deem, and C. F. Lucks, June 12, 1961.

* Superheat Advance Demonstration Experiment loop in the VBWR. This loop has been expanded and modified to the E-SADE loop.

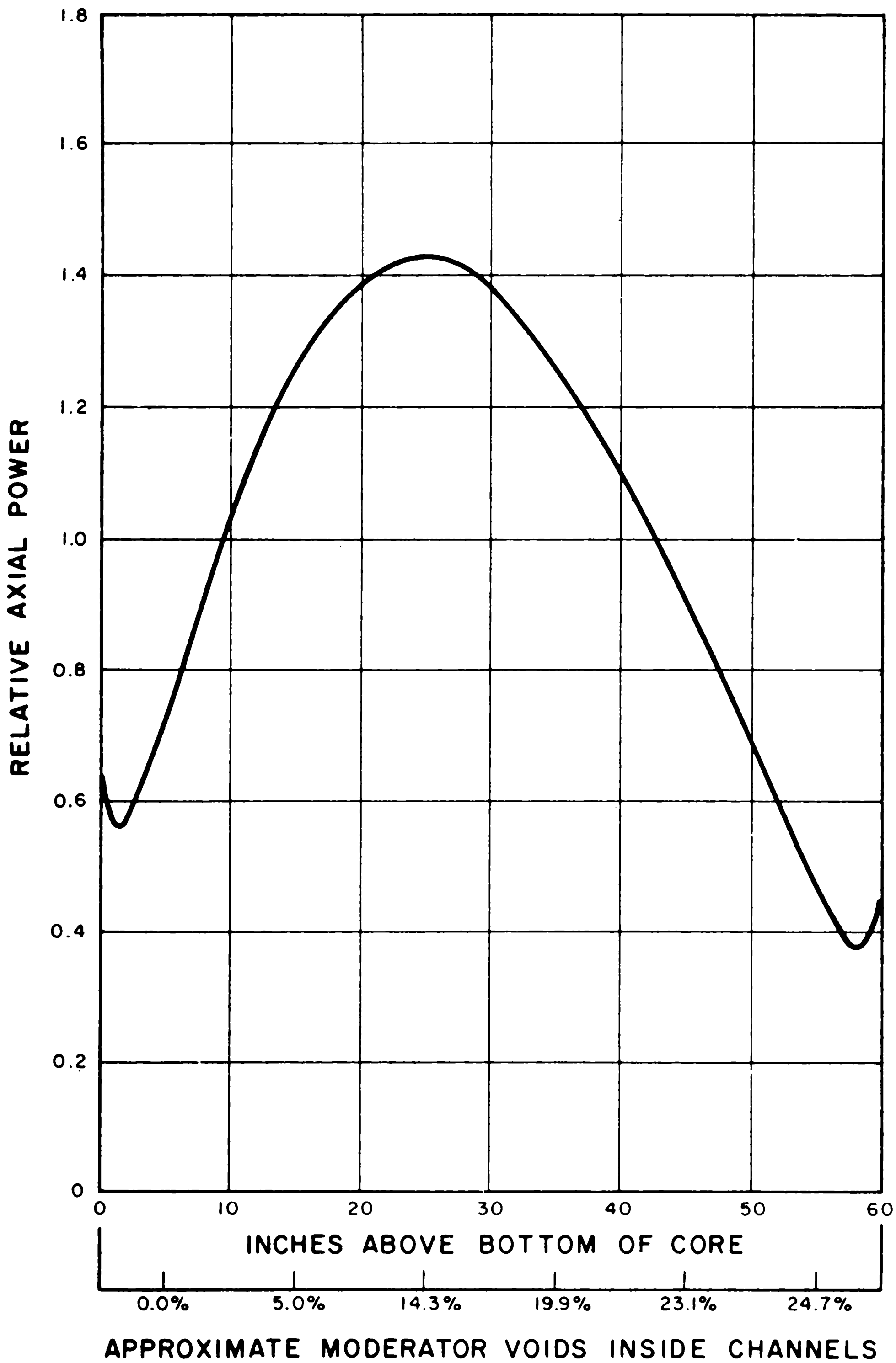

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FIGURE 4.7
AXIAL POWER DISTRIBUTION
FULL POWER, BEGINNING OF LIFE

the UO_2 temperature range of interest for the Mark II core and lies in the flat minimum zone given in the reference. The exact value of the thermal conductivity is relatively unimportant due to the low UO_2 operating temperatures in EVESR.

The pressure drop across each of the 9 elements in a given fuel bundle is the same. Since the heat generation varies among the 9 elements in the single bundle (Figure 4.8), resulting steam density differences affect the individual pressure drops and different flows result in each element within a fuel bundle as shown by Figure 4.6. The flow through each bundle is set by the flow required by the hot element. The other fuel elements in each bundle, then, are overcooled.

The temperature shown for each fuel element on Figure 4.6 is the "most probable temperature" present. The "most probable temperature" is the peak circumferential average temperature plus the temperature rise caused by local circumferential flux skewing. The peak circumferential average temperature includes the overall axial and radial factors only.

The temperature rise caused by local circumferential flux skewing was analyzed in detail. A typical plot of the local neutron flux around an element is shown on Figure 4.9. This plot would be of interest in determining the effect of skewing on the outer cladding temperatures. Figure 4.10 shows the temperature rise (or drop) in the steam and cladding for various amounts of flux skewing for the range of skewing expected over the entire core (0 to 0.33). This flux skewing temperature change must be added to or subtracted from the circumferential average clad temperature, depending on whether the temperature being calculated

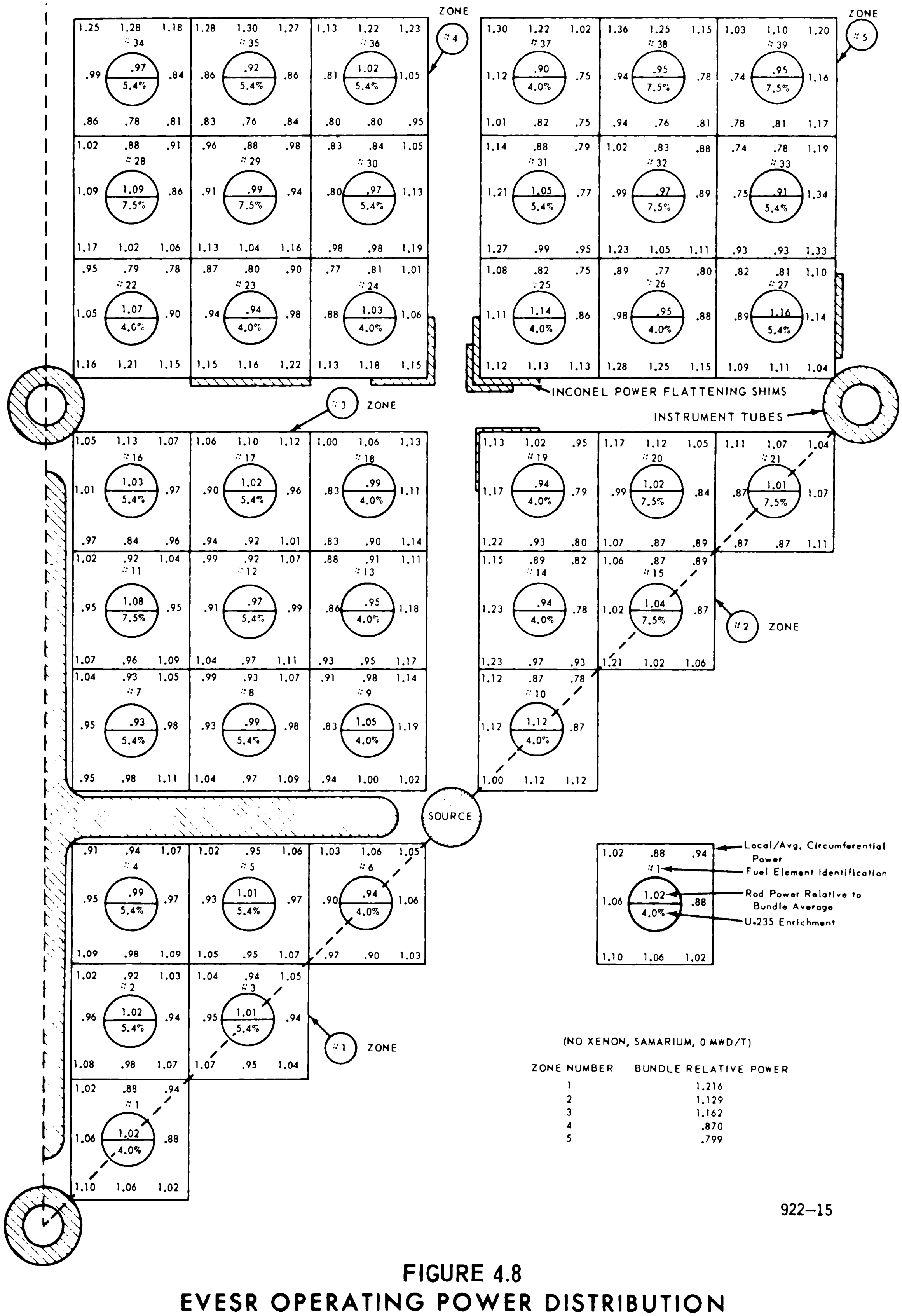

FIGURE 4.8
EVESR OPERATING POWER DISTRIBUTION

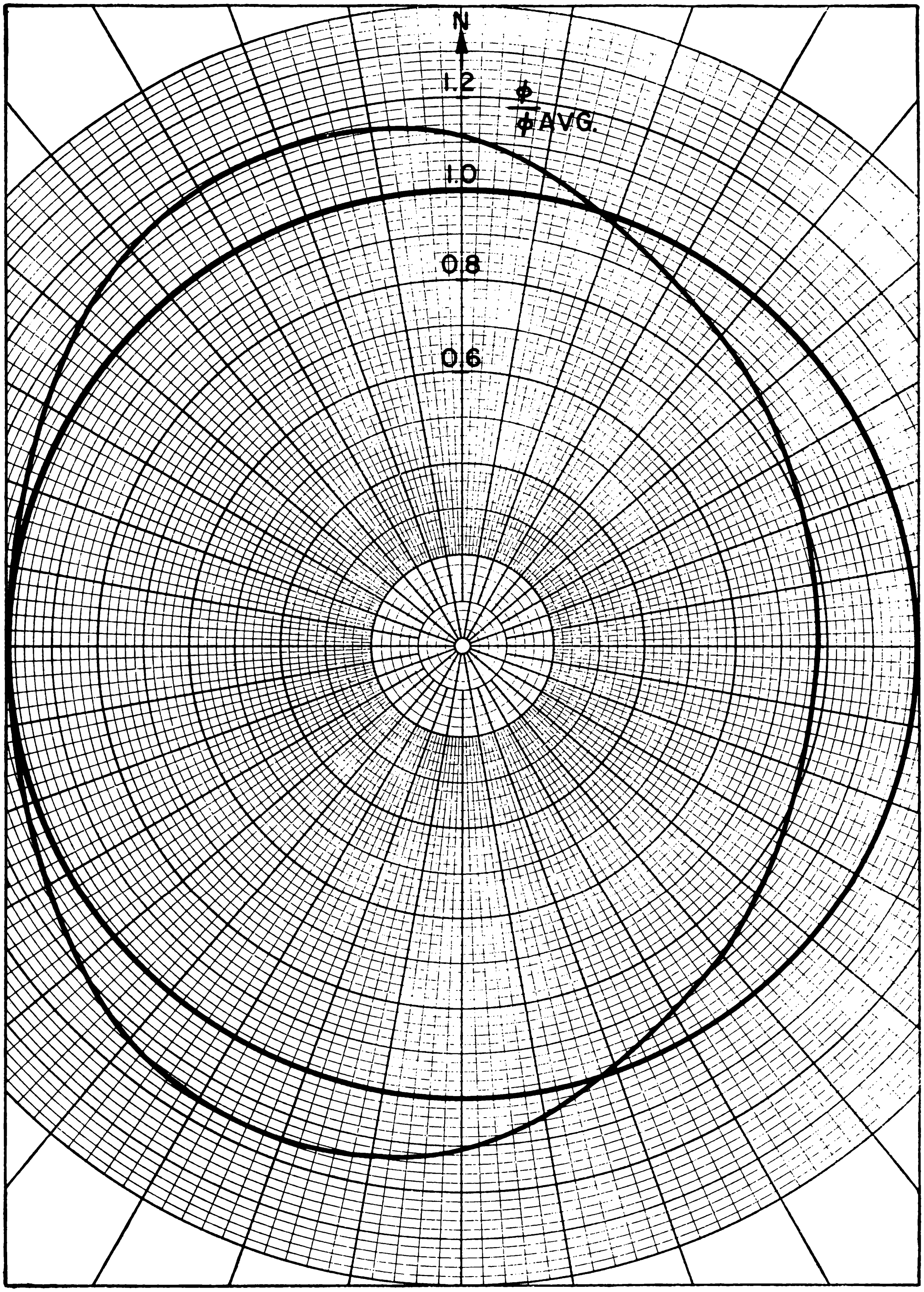

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FIGURE 4.9

TYPICAL LOCAL FLUX PERTURBATION AROUND FUEL ELEMENT (HOT ELEMENT)

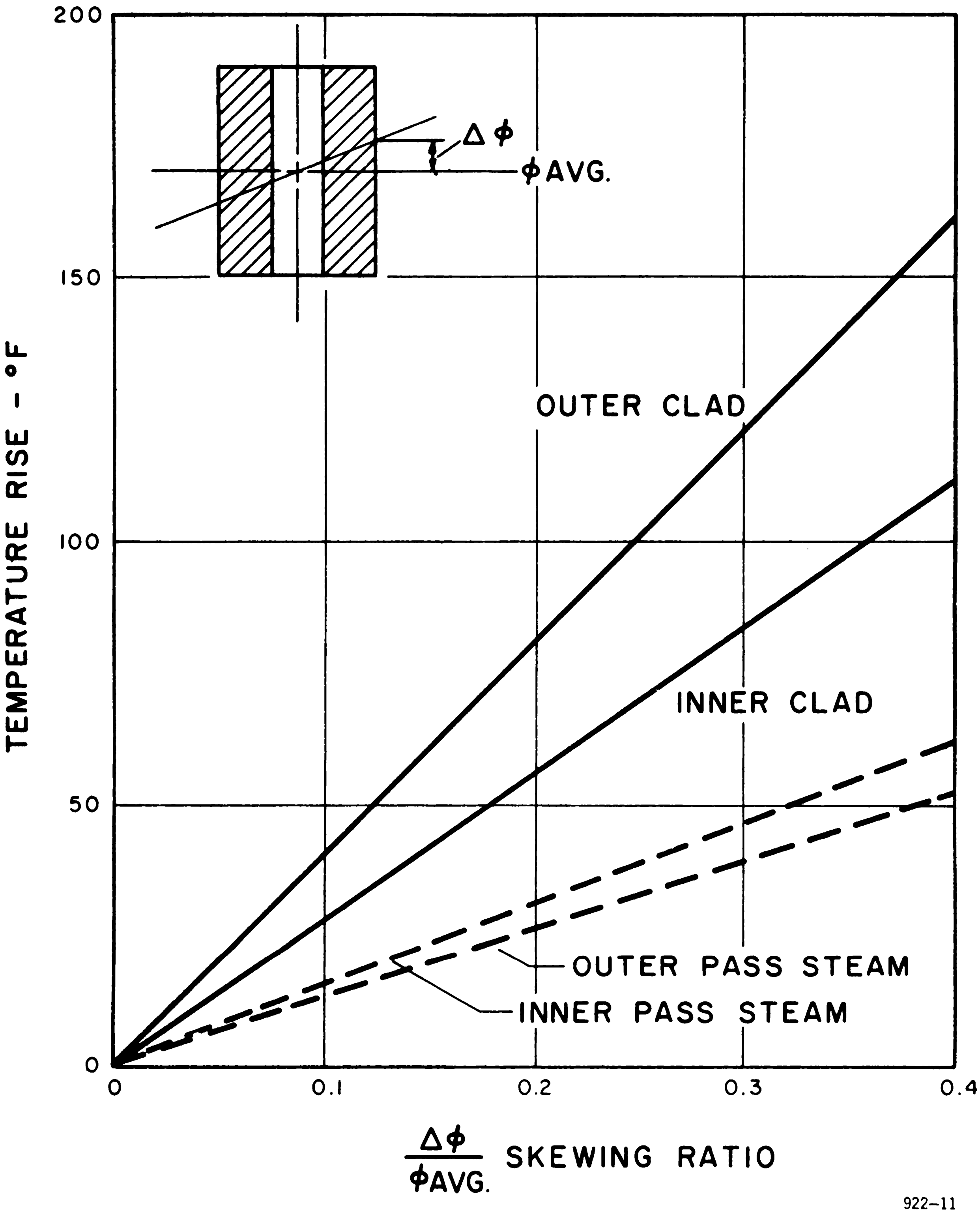

FIGURE 4.10
EFFECT OF FLUX SKEWING ON CLAD AND STEAM TEMPERATURES AT HOT SPOT

is on the high or low side of the flux. Thus, the "most probable temperature" includes all factors known to be present, but excludes the effects of statistical variations or uncertainty factors. Since the uncertainties which cause a departure from the most probable temperature are random, their effects on clad temperature were combined statistically to give an overall temperature uncertainty associated with the most probable temperature. The "maximum clad temperature" is equal to the "most probable temperature" plus or minus the overall statistical temperature uncertainty.

As an example, in the hot element in Zone II, the "circumferential average temperature" is 1043°F, which includes axial and radial factors. The "most probable temperature" is 57°F higher due to local flux skewing, or 1100°F. The uncertainty in this temperature, found by combining the various uncertainties statistically, is plus or minus 150°F. Since the various uncertainties were taken at 3 sigma limits, the probability that the maximum clad temperature lies within this overall temperature uncertainty is 99.7 percent, i.e., in 997 out of 1000 cases, the "Maximum temperature" should be no higher than 1250°F, nor lower than 950°F.

The uncertainty factors included in calculating the "maximum clad temperature" are as follows:

1. Uncertainties in heat generation determined from physics calculations, as well as variation in fuel density, fuel pellet sizes and enrichment.

2. Flow rate uncertainties for flow maldistribution among bundles due to inlet conditions, flow maldistribution within each bundle, and orificing, flow meter, and thermocouple errors.

3. Steam film drop factors to account for heat transfer coefficient data scatter; local mechanical variations of the flow annulus as determined from fuel bowing tests; local error due to eccentricity of the fuel element within the process tube, calculated from tubing tolerances; uncertainty in estimating the inside clad to fuel contact coefficient; uncertainty in estimating the heat loss through the process tube; uncertainty in the UO_2 conductivity; and local pellet size, pellet density and enrichment outside the normal manufacturing tolerances.

The axial temperature variation of the hot element is shown on Figure 4.11. The radial profile at the hot spot is given on Figure 4.12.

As the inner control rods are withdrawn later in core life, the relative bundle power generations will shift. It is planned to perform analyses similar to that indicated above in this section at appropriate intervals and make adjustments of the required bundle flows. Sufficient steam flow is available to cool the core adequately throughout all planned rod patterns. Flux wire and exit steam temperature measurements are planned for at least 1/8 of the start-up core. Adjustments in the calculated steam flow rates will be made, if appropriate, to reflect the results of these measurements.

The cladding temperatures shown in Table 4.1 and on Figure 4.6 are for steady-state operation at 12.5 MW(t). A specific overpower allowance for such things as maneuvering and hunting has not been included in these cladding temperatures or in the uncertainty factors since the rated power

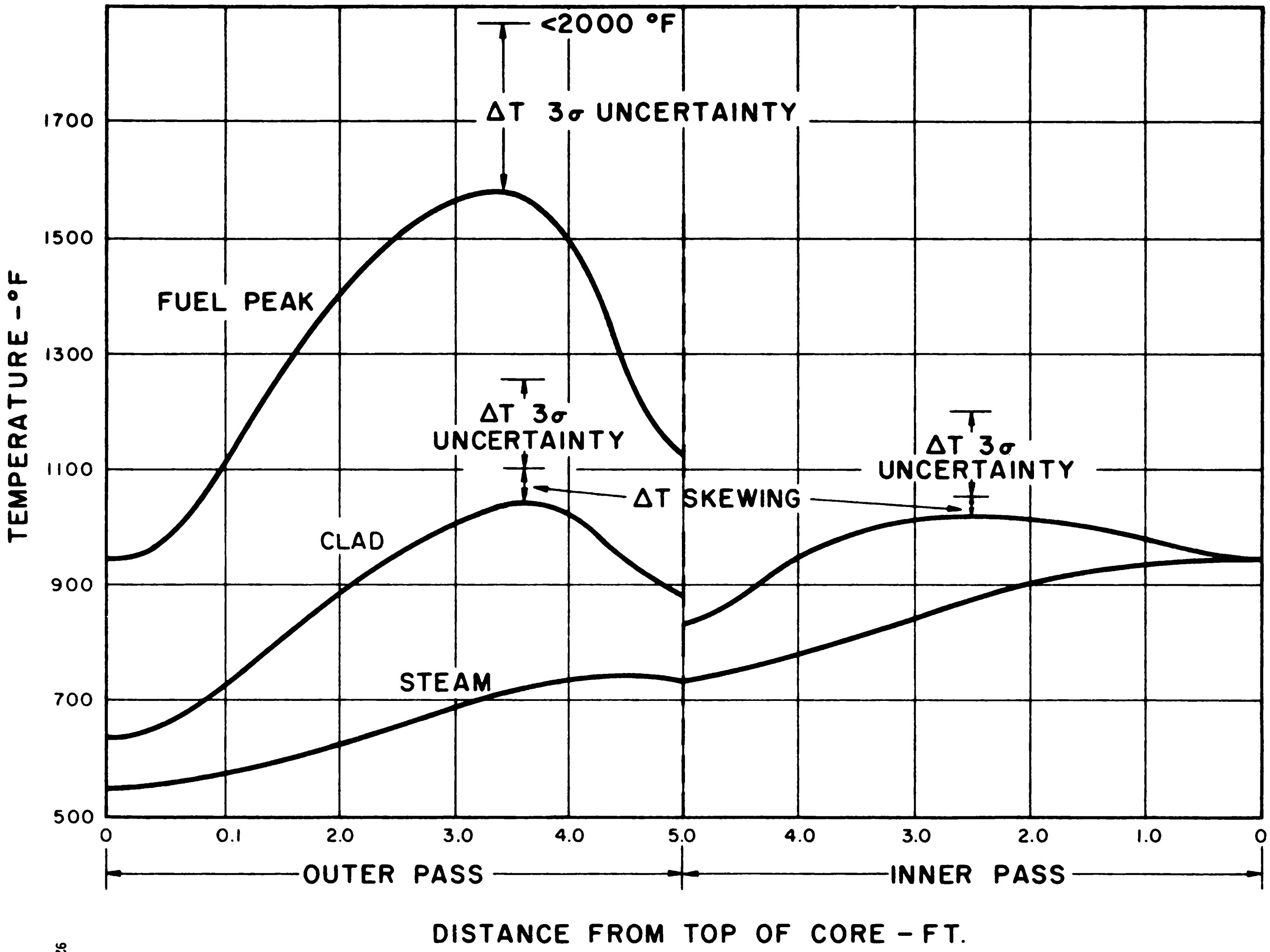

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FIGURE 4.11

HOT ELEMENT AXIAL TEMPERATURE PROFILE

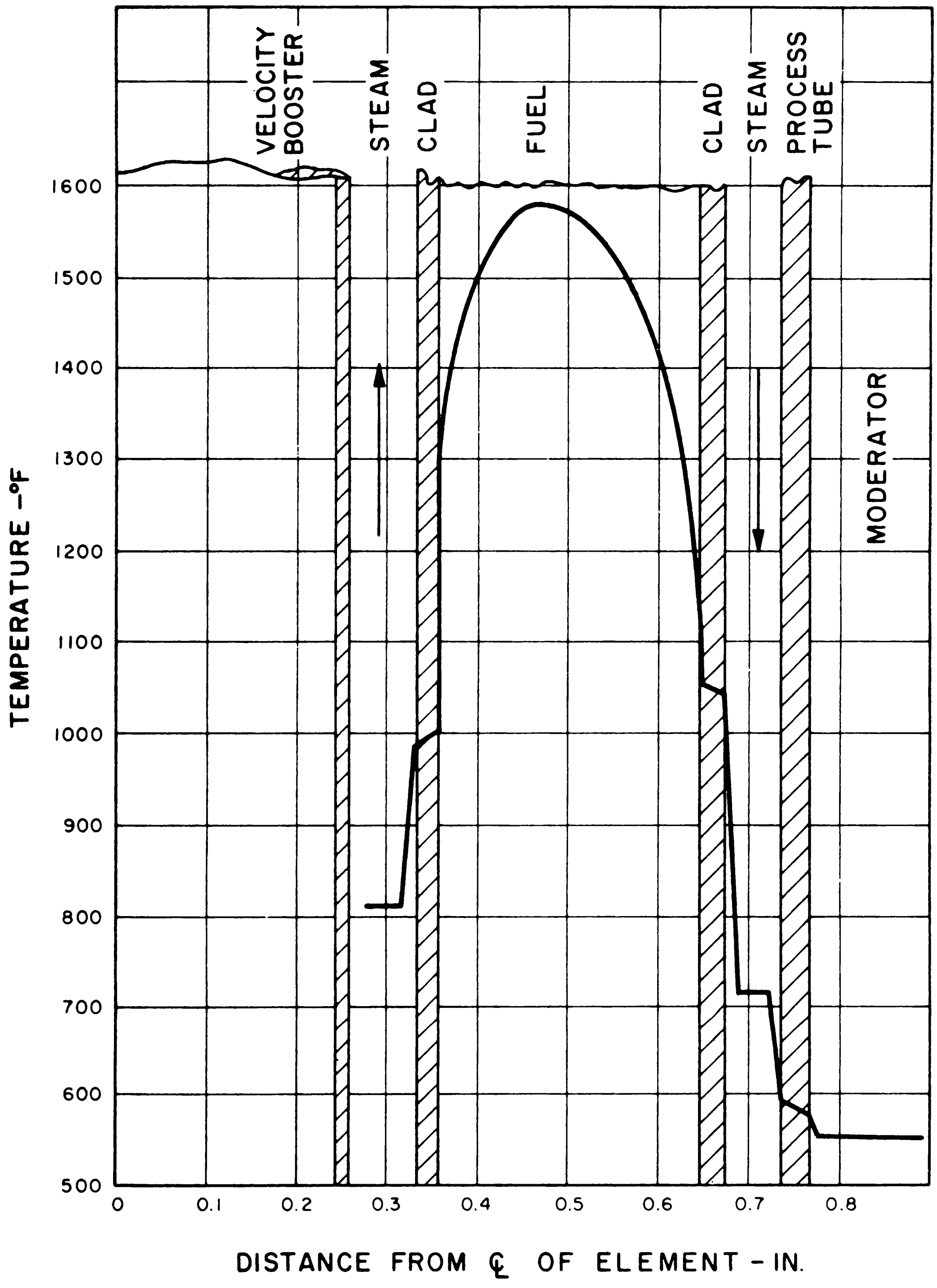


FIGURE 4.12
HOT ELEMENT RADIAL TEMPERATURE PROFILE

maximum temperature of 1250°F is considered a nominal maximum value around which to operate, rather than a "hard" limit which should not be exceeded except in accident situations.

With the steam flow control system operating properly, the clad temperatures and other pertinent parameters as a function of steady-state core power level are shown on Figure 4.13. At 10 percent overpower, the neutron flux alarm setting, the clad temperature has increased 30°F, the steam temperature 20°F, and the UO_2 temperature 80°F, above the normal full power values. The neutron flux scram setting will be at 25 percent overpower. The hot element clad temperature rise above normal would be 75°F, the hot bundle steam temperature rise 50°F, and the UO_2 temperature rise 200°F at this steady-state overpower condition.

The maximum allowable bundle outlet steam temperatures are determined by the design temperatures of the exit steam piping. The design temperature of the piping from the four central fuel bundle locations is 1050°F and for all others 950°F. Hence, the maximum outlet steam temperature alarm settings must not exceed 1050°F and 950°F, or lower than this if the clad temperature is limiting before the piping temperature. In the Mark II core, if it is assumed that the alarm trips are set 30°F above the estimated bundle outlet temperature, it is seen from Figure 4.13 that the alarm would trip at 15 percent overpower. This results in a 50°F rise above normal in the hot element clad temperature, and 120°F in the UO_2.

The nuclear characteristics of the core make the possibility of local power variations affecting only one bundle highly improbable. In any event, even if a bundle outlet temperature alarm at 30°F above normal

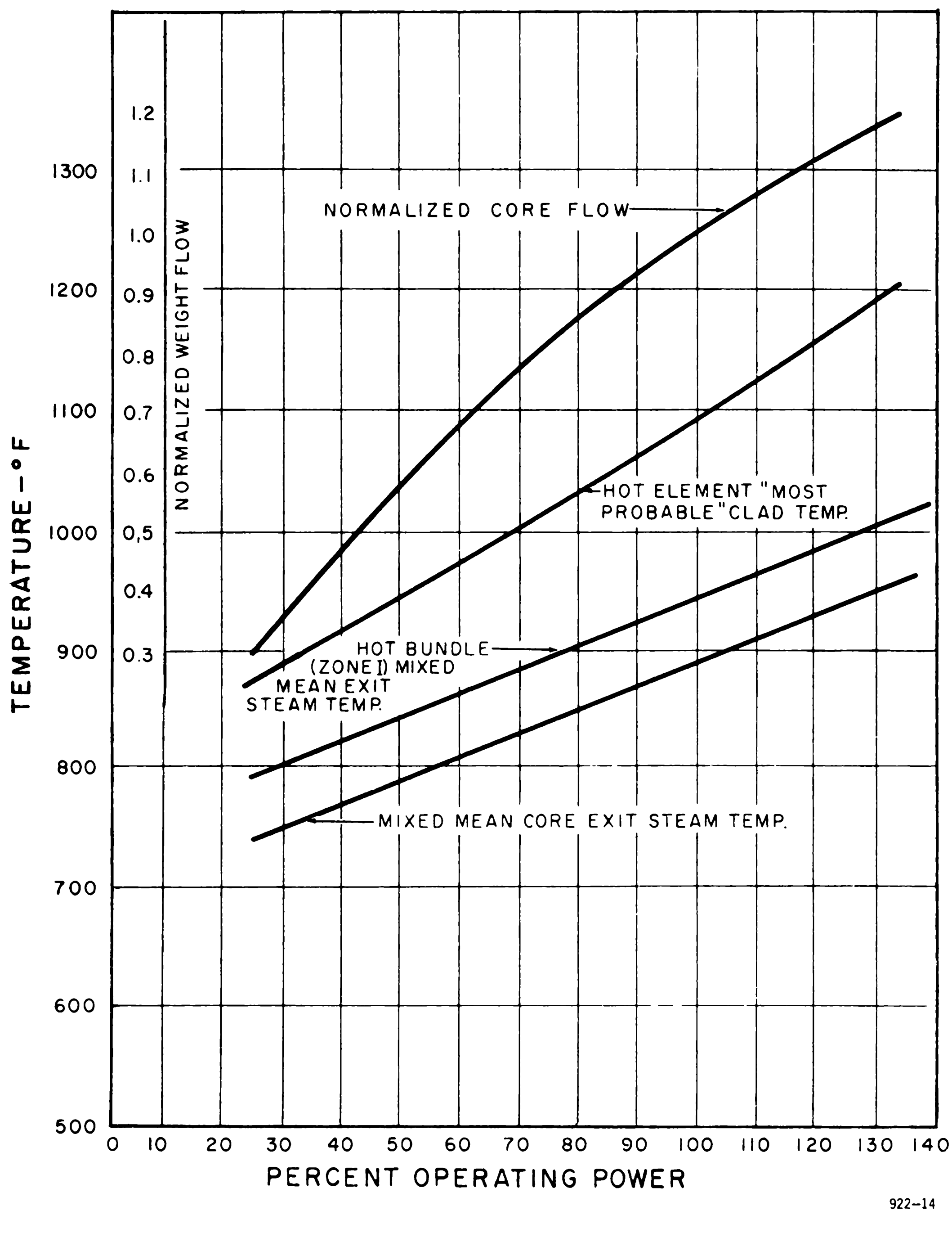


FIGURE 4.13

PART-LOAD TEMPERATURE AND FLOW CHARACTERISTICS

(SUPERHEATED STEAM REGION)

should trip due to local high power while the reactor as a whole is running normally at full power, the clad temperature rise in the hot element would only be $40^{o}F$. This would represent a 5 percent increase in local bundle power. The difference between this case and the case discussed in the paragraph above is that the control system would not provide flow compensation for a local high power situation.

If the control system were to malfunction so that flow compensation for temperature did not occur, and the reactor power were to increase slowly, the situation would be similar to the local power increase discussed in the paragraph above. The core mixed mean steam outlet temperature, as well as individual fuel bundle steam temperatures, would alarm ($30^{o}F$ above normal) at 5 percent overpower and the hot element clad temperature rise would be $40^{o}F$ at that time.

Should the reactor be running at normal power level, but the control system somehow malfunction such that the core steam flow rate were in error, the high temperature alarm on exit temperature would sound when the flow drop was 7 percent. The cladding temperature would increase $40^{o}F$ and the UO_2 temperature less than $50^{o}F$.

The mechanisms for failure of superheat fuel are not well defined at this time. Experiments and analytical work point to interdependence between incompletely understood corrosion mechanisms, and complex steady-state or cyclic temperature-stress-time relationships in the fuel element cladding and fuel. No one parameter may be adequate in defining an upper limit. One of the purposes of building the EVESR is to determine such limits. The present cladding design temperature of 1250°F and less than 15 KW/Kg specific power maximum are not near any known hard limit. As clearly illustrated by the SADE tests on similar fuel, sudden failure did not occur even at prolonged operation at surface temperature 250°F in excess of 1250°F and at specific power levels nearly three times as high as in the EVESR hot element.[1] As nearly as current knowledge implies, the principal effect of exceeding the design conditions for short periods from any cause is expected to be a decrease in the long-term life of the fuel element. This is why 1250°F is not considered to be an upper limit in normal operation of the reactor, as brought out earlier. From the corrosion standpoint, it is known that chloride stress corrosion has been a factor in the failure of stainless steel cladding in SADE loop experiments, but experience indicates this is not a catastrophic type of failure.

For the cladding, the more severe types of transients are expected to be those which might occur from such events as scramming from full power, emergency cooling initiation, and flooding of the core too early after shutdown, rather than small increases above the normal operating temperature. In such cases, the cladding could suffer abnormal thermal strains because of its fast response relative to the fuel and end closures, or longitudinal differential expansion between the inner and outer cladding. The steam flow control system is designed to prevent severe shocks following

1. GEAP-4024, Nuclear Superheat Project Tenth Quarterly Progress Report, October - December 1961, Contract AT(04-3)-189, P.A. #13, page 30.

a scram by the use of the steam temperature trim which resets the steam flow as core power changes. The emergency cooling system is designed to keep the cladding temperature rise within 200°F even for complete loss of flow. The flooding procedures are designed so that the flooding temperature differential between the cladding and the water should be about 50°F maximum. In a qualitative test, three-foot test sections were heated to 800°F and inserted in 100°F water such that the outer cladding was suddenly cooled relative to the inner clad and end closures. Five cycles failed to reveal any gross distortions, and the element remained leak-tight. It is not expected that the types of transients discussed above will in themselves cause clad rupture. Repeated cycling of this kind could shorten the life of the fuel element and eventually lead to fissures with subsequent release of gaseous fission products as in the SADE experiments.

The UO_2 fuel temperature is not a limiting factor in the design for steady-state operation of the Mark II core. Calculations indicate that the maximum UO_2 steady-state temperature will be below 2000°F, as compared with the melting point for UO_2 of about 5000°F, and the UO_2 recrystallization temperature of approximately 3000°F.

4.3.3 Moderator Region

The uninsulated process tubes of the Mark II fuel receive heat by convection from the superheated steam in the first pass, and by radiation from the outer fuel cladding. This heat is dissipated to the moderator by convection and boiling. The process tube remains at essentially the moderator temperature because of the low thermal impedance on the water side of the process tube compared to that on the steam side.

The heat loss by convection from the superheated steam amounts to 35.7 percent of the total reactor power. In addition about 5 percent of the total reactor power is generated directly in the moderator and the structural materials. The axial distribution of the heat loss to the moderator is shown on Figure 4.14. The total heat loss is 40.7 percent for the 12.5 MW(t) core. It is anticipated that in the future the reactor will be used to test fuel having insulated process tubes. This would reduce the heat losses to something under 10 percent.

The steam-water mixture which forms in the core channels is lighter than the corresponding water leg between the vessel wall and the core shroud. This unbalance causes an upward recirculation through the core and down around the outside of the core shroud at a rate determined by the total steam being formed and the hydraulic resistance of the complete loop. About 90 percent of the resistance is in the two-phase portion of the loop. A shroud surrounding the core extends 2 feet above the core to provide additional driving head to assist recirculation.

The recirculation rates and axial void distribution in each of the 5 thermal zones were determined by means of a digital computer code used in the design of boiling water reactors. The axial void distribution through each of the 5 zones is shown on Figure 4.15. Pertinent data for each of the zones is tabulated and a summary of the data of overall moderator region is given in Table 4.2.

At the top of the shroud, the steam begins to disengage from the water and rises to the top of the vessel where it mixes with the incoming steam

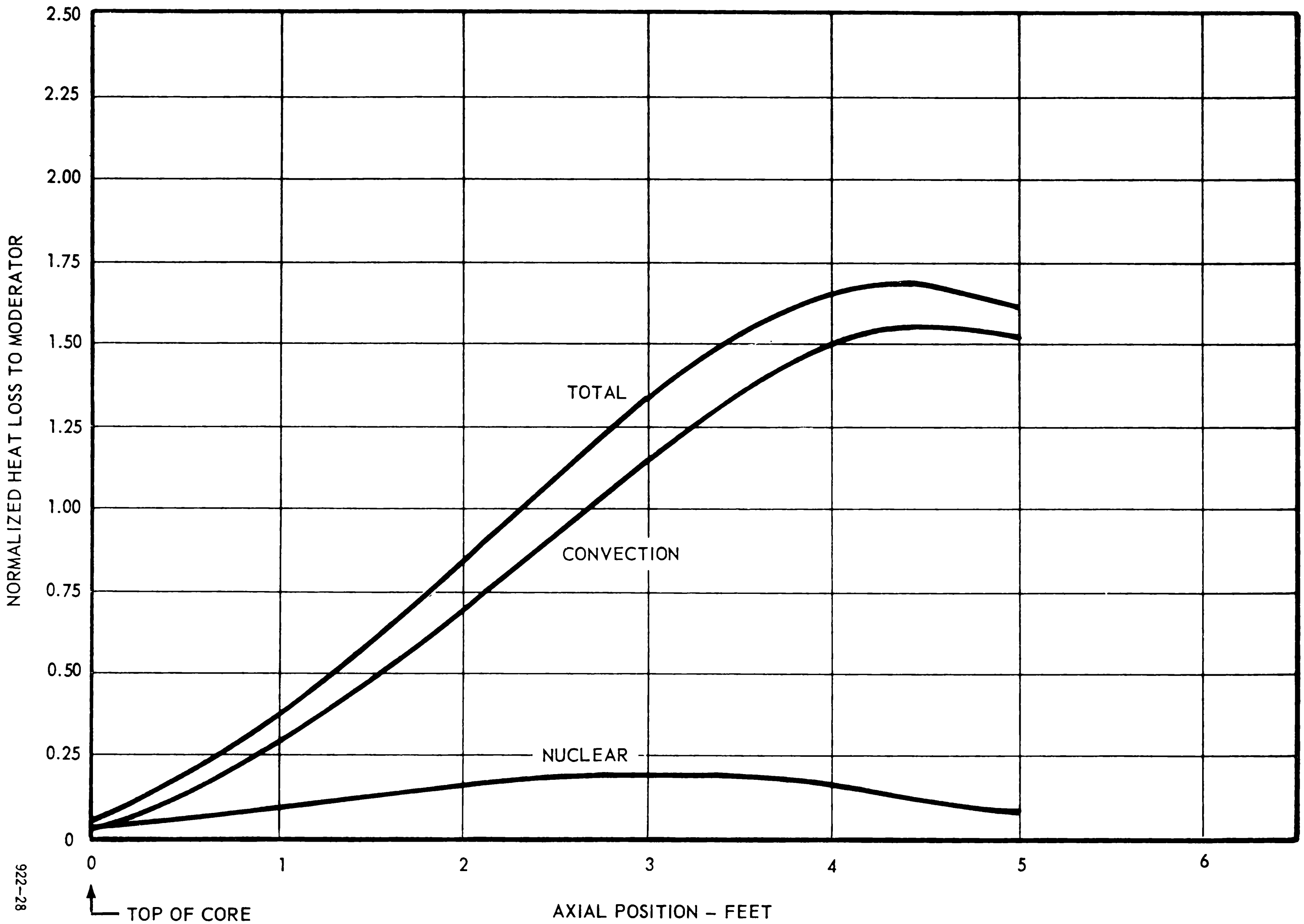


FIGURE 4.14
HEAT LOSS TO MODERATOR
(MARK II CORE HOT ELEMENT)

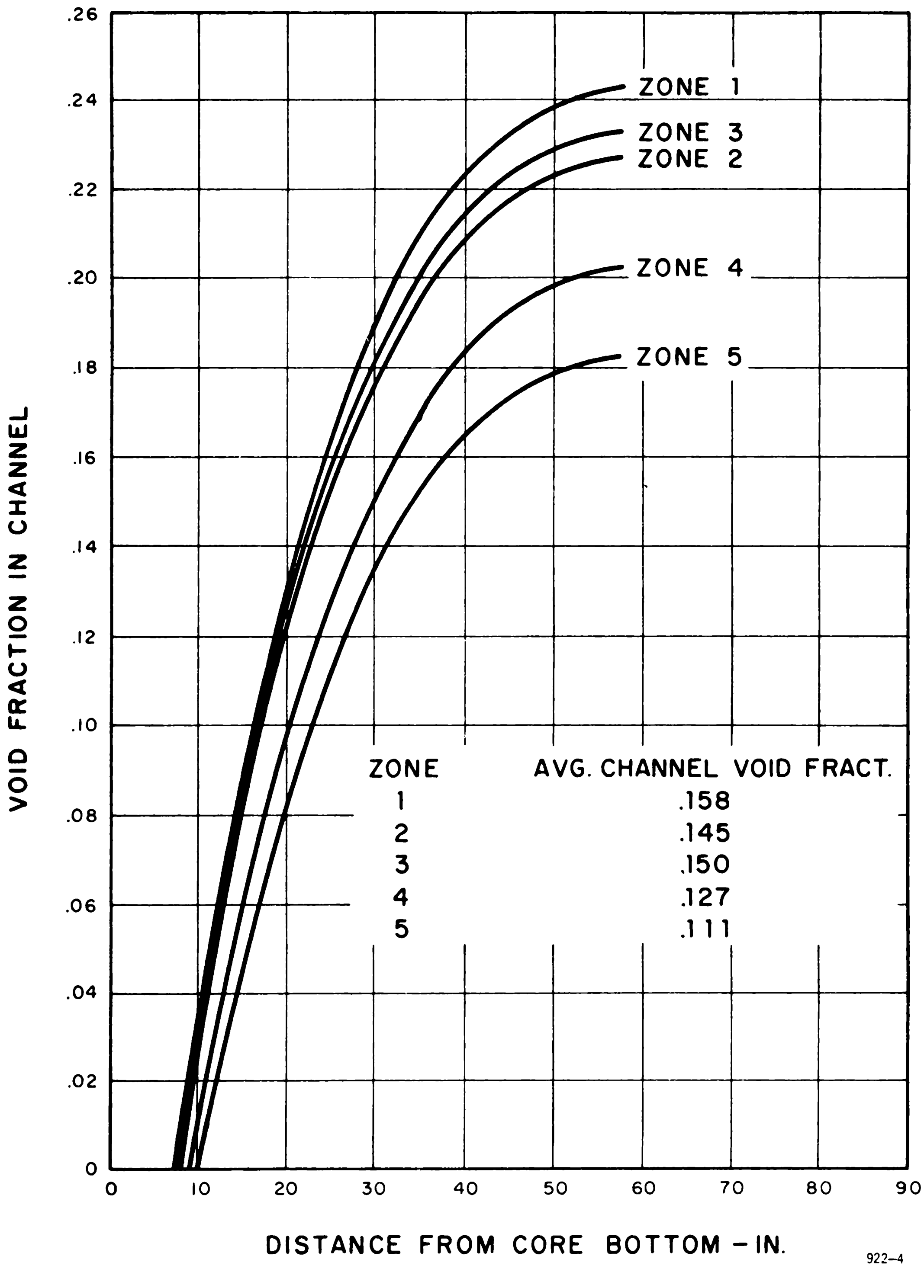


FIGURE 4.15
AXIAL VOID DISTRIBUTION
(12.5 MWT OPERATION)

Table 4.2

Moderator Data

Overall Moderator Data

Power generated in water (% of reactor power)	%	3.46
Power generated in process tubes	%	0.35
Power generated in control rods	%	0.34
Power generated in channels	%	0.46
Power generated in miscellaneous structures	%	0.39
Total power direct to moderator	%	5.0
Total power generated within moderator (5%)	Btu/hr	2,133,000
Power transferred to moderator from process tubes	Btu/hr	15,228,000
Makeup water inlet temperature	^{o}F	130
Core inlet subcooling	Btu/lb	3.5
Downcomer mass flow	lb/hr	2,144,000
Downcomer velocity	fps	0.49
Total mass flow through fuel channels	lb/hr	1,669,300
Total mass flow through control rod channels	lb/hr	259,200
Total mass flow bypassed	lb/hr	215,500
Average core exit voids	Vol. %	11.8
Overall core average voids	Vol. %	7.1
Free surface steam leaving velocity	fps	0.1

Thermal Zone Data	Zone I	Zone II	Zone III	Zone IV	Zone V
Total flow rate, lb/hr	53,570	52,700	53,040	52,460	50,030
Exit channel voids, % by vol.	24.4	22.8	23.4	20.1	18.3
Avg. channel voids, % by vol.	15.8	14.5	15.0	12.7	11.1
Boiling length, fraction	0.89	0.87	0.88	0.85	0.83

from the gas-fired boiler. The steam leaving the free surface has a velocity of less than 0.1 fps. Tests indicate that free surface separation can be obtained at velocities of nearly 1 fps with a wetness under 0.2 percent, two feet above the indicated water level. The steam downcomer inlets are 4-1/2 feet above the high normal indicated water level. The steam from the gas-fired boiler passes through a dryer before it reaches the reactor, which guarantees that this steam will have a quality below 0.1 percent. Hence, the net wetness of the steam entering the steam downcomers of the fuel bundles should be under 0.2 percent. If credit is taken for the heat losses from the superheated steam jumpers to the steam in the reactor vessel, the steam actually entering the downcomers should be essentially dry. Additional drying will take place in the downcomers due to heat losses from the risers.

As a function of steady-state reactor power, the recirculation rates and other pertinent parameters of the moderator are shown on Figure 4.16. Provisions for orificing of all the fuel and control rod channels have been made for future use. The heat generation in the control rods is small and no net voids are produced in the control rod channels. It should be noted that the core void fractions are well below values normally associated with proven boiling water reactors.

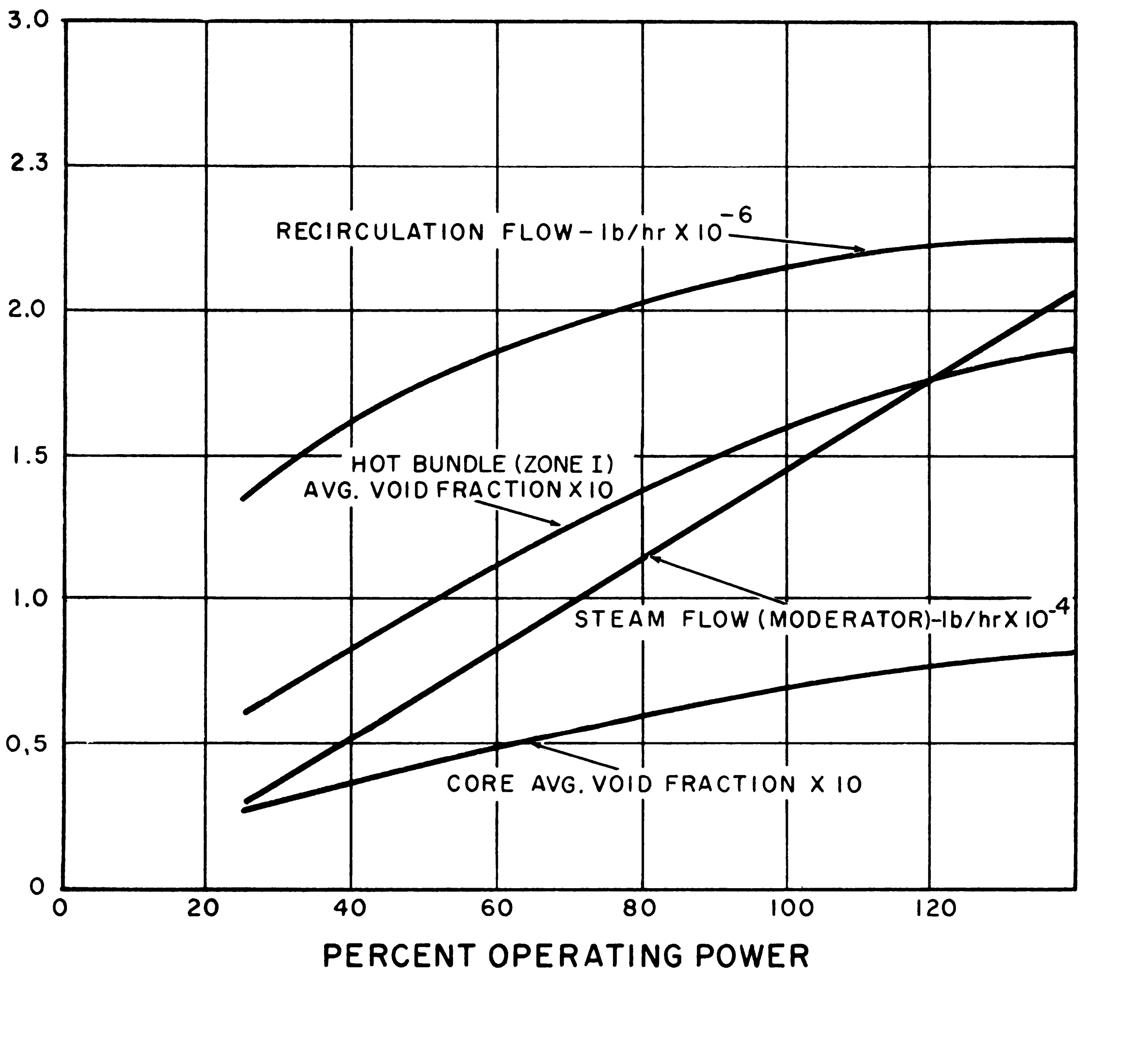


FIGURE 4.16

PART-LOAD MODERATOR CHARACTERISTICS

5.0 SUB-TASK B-2 FUEL DESIGN ENGINEERING PHYSICS

5.1 General Nuclear Characteristics of Mark II Core

The EVESR, being a light water moderated, steam-cooled reactor, possesses most of the nuclear characteristics of a boiling water reactor with the heat transfer characteristics of a gas-cooled reactor. Nuclear characteristics affecting the core reactivity, reactivity coefficients, and power distribution will be discussed below. While these parameters were determined primarily by analytic methods, critical experiments on a detailed engineering mock-up of the reactor have been used to confirm and to make minor adjustments to the analytic values.

Table 5.1 shows the volume fractions of various materials present in the core with the superheated steam passages flooded with water and unflooded, and with the control rods fully out and fully inserted.

From the nuclear viewpoint, the following are the significant physical characteristics of the start-up (Mark II) core:

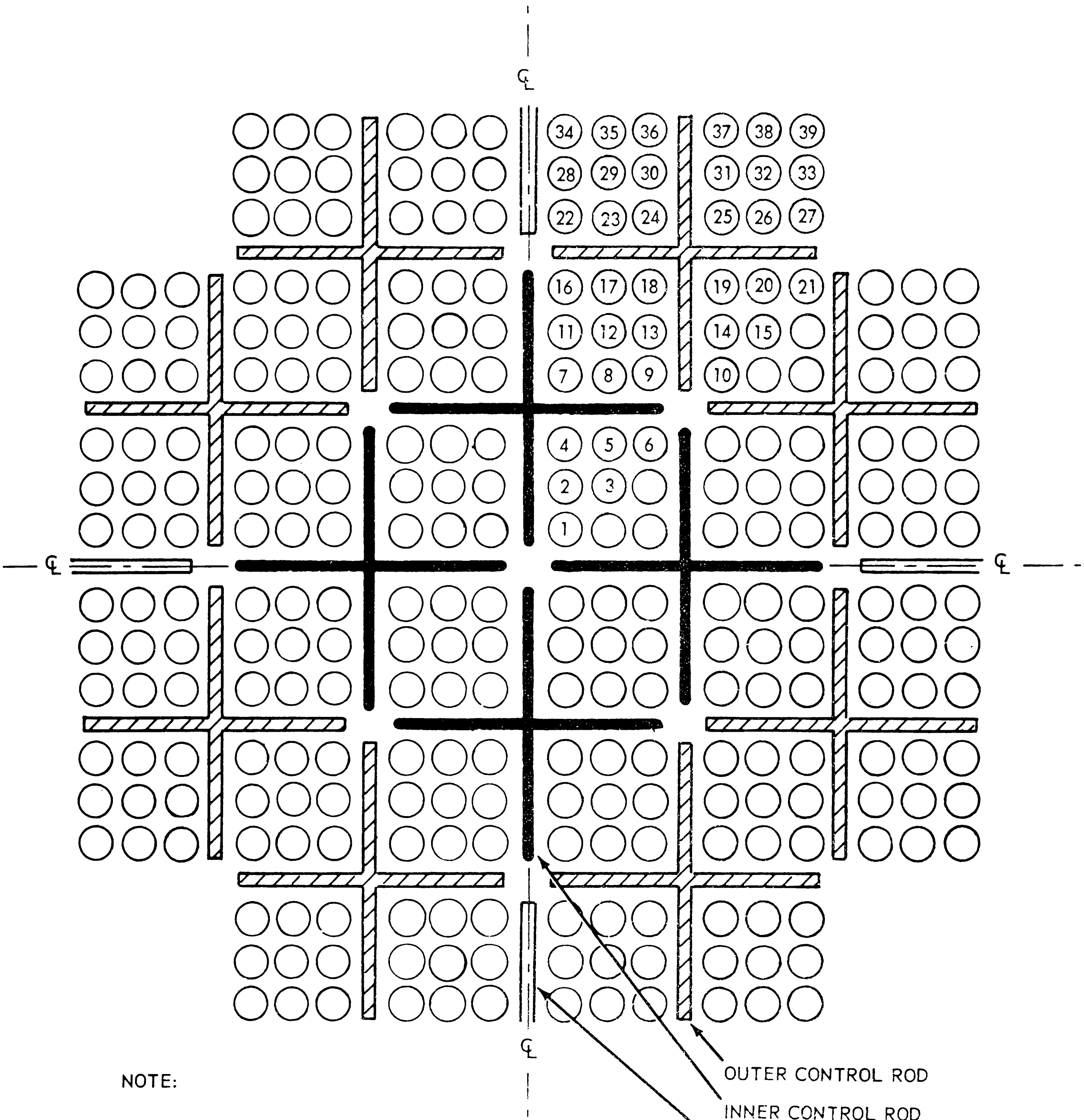
1. The core consists of 32 fuel bundles arranged in a 6 x 6 array with the corner positions vacant as shown on Figure 5.1. The active fuel length is 60 inches and the equivalent core diameter is 42.1 inches.

2. The fuel bundles are comprised of 9 annular fuel elements in a 3 x 3 square array surrounded by a zircaloy channel. The control rods operate within a 3/4 inch water gap between zircaloy channels as indicated on Figure 5.2.

Table 5.1

Material Volume Fractions in the Mark II Core

	Flooded		Unflooded	
	Rods Out	Rods In	Rods Out	Rods In
Uranium Dioxide	0.1795	0.1795	0.1795	0.1795
Water	0.7009	0.6123	0.5778	0.4892
Stainless Steel (Velocity Booster, Process Tube, and Misc.)	0.0448	0.0448	0.0448	0.0448
Zircaloy Channel	0.0337	0.0337	0.0337	0.0337
Fuel Clad and Wire Spacers	0.0398	0.0398	0.0398	0.0398
Main Control Rods	0	0.0886	0	0.0886
Superheated Steam Passages				
Outer Annulus	0	0	0.0603	0.0603
Inner Annulus	0	0	0.0265	0.0265
Void (Inside Velocity Booster, Fuel Clad Gaps)	0.0013	0.0013	0.0376	0.0376
Water-to-Fuel Volume Ratio	3.9	3.4	3.2	2.7



NOTE:

NUMBERS INDICATE THE
39 UNIQUE FUEL ELEMENTS
FOR 1/8–CORE SYMMETRY

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FIGURE 5.1

EVESR CORE HORIZONTAL CROSS SECTION

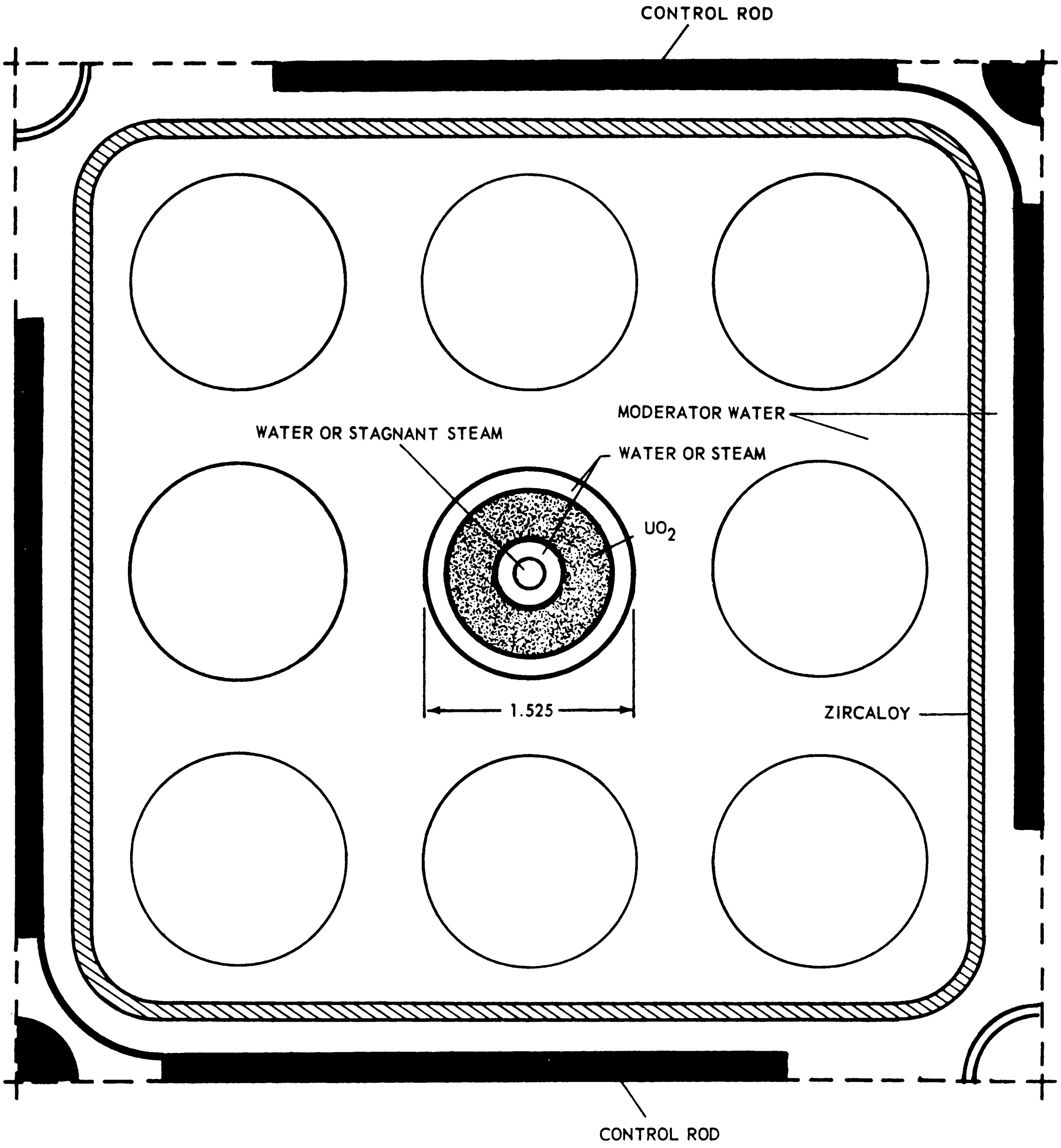

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FIGURE 5.2
EVESR 9-ROD FUEL BUNDLE

3. The fuel material is uranium dioxide of 93.5 percent theoretical density of three different enrichments: 4.0, 5.4, and 7.5 percent U-235 by weight. The core average enrichment is approximately 5.4 percent U-235 by weight. The uranium dioxide pellets are clad with various types of stainless steel, Incoloy, or Inconel, depending upon core position. Under normal operating conditions, the volumetric average uranium dioxide temperature is about 1250°F, and the average surface temperature is 1000°F.

4. The moderator is light water. Under normal operating conditions (nominal 1000 psig, 545°F), there is steam formation in the upper portion of the core. Steam "voids" are normally confined to the moderator enclosed by the zircaloy channels, and range up to approximately 27 percent of the moderator volume at the exit of the hottest channel. The overall core average void content at full power is about 14 percent by volume.

5. The initial core is designed for an average burnup life of 4000 MWD/T of uranium. The enrichment has been set to achieve this exposure with sufficient reactivity to operate at 12.5 MW(t) rated power with equilibrium zenon and samarium at the end of life. The initial clean core multiplication with all control rods withdrawn will be approximately 1.08 at full power.

6. Reactivity control is achieved with 12 main cruciform control rods and 4 peripheral scram rods arranged as shown on Figure 5.1. The main control system is capable of controlling a reactivity change of 13 percent Δk in the cold, flooded conditions and 19 percent Δk in

the hot unflooded condition with voids. The 4-rod peripheral scram worth is a maximum of about 2.0 percent Δk.

The nuclear characteristics of the fuel element and fuel bundle designs have been carefully optimized with respect to the overall plant safety. From the beginning of the design effort, it was recognized that an optimization of the water-to-fuel volume ratio was of prime importance in establishing a safe design, since this parameter has a direct influence on the magnitude of the reactivity change upon flooding or unflooding the steam coolant passages with water. This optimization recognized the possibility of two extreme reactor conditions, both of which yield positive reactivity additions:

1. Accidental unflooding, in which water is inadvertently ejected from the steam coolant passages in the core when the moderator is cold; and

2. Accidental flooding, in which water is introduced into the steam coolant passages when the moderator is hot and pressurized.

To determine the proper design point with respect to the reactivity changes due to these two extreme mechanisms, it was necessary to consider the entire spectrum of potential nuclear accidents and the effect of a change in the nuclear design on each. Since unflooding is inherently autocatalytic, since it can occur rapidly, and since when it is minimized the maximum positive void and temperature defects are small, it received more weight than the hot flooding in the nuclear optimization for safety. It is very significant that the nuclear optimization discussed above was

Table 5.2

Lattice Parameters
(Control Rods Out)

	Cold, $68^{o}F$		Hot, $545^{o}F$		
Parameter	Unflooded	Flooded	Unflooded	Flooded	Unflooded, 14% Voids
r	41	30	63	47	79
L^2	3.2	2.7	6.0	5.0	6.5
M^2	44	33	69	52	85
$p^{(U-238)}$	0.92	0.93	0.89	0.90	0.88
k_∞	1.18	1.15	1.25	1.23	1.25
B_m^2	0.0024	0.0021	0.0020	0.0021	0.0019
k_{eff}	1.07	1.08	1.10	1.12	1.08
ℓ	0.4×10^{-4}	--	0.3×10^{-4}	--	--
β	--	--	0.007	--	--

Legend

r = Fermi Age, cm^2

L^2 = Thermal Diffusion Area, cm^2

M^2 = Migration Area, cm^2

$p^{(U-238)}$ = Resonance Escape Probability, U-238

k_∞ = Infinite Multiplication Constant

B_m^2 = Material Buckling, cm^{-2}

k_{eff} = Effective Multiplication Constant

ℓ = Neutron Lifetime, sec.

β = Effective Delayed Neutron Fraction based primarily on safety considerations and not on economics or other factors. The moderator (water)-to-fuel volume ratio chosen was 3.2 (with superheat coolant passages unflooded, control rods withdrawn).

Additional significant nuclear lattice parameters are presented in Table 5.2 for the control-free condition.

5.2 Physics Calculational Model

The physics calculational model used in the analyses is the synthesis of several individual models. Its accuracy in predicting reactivity, reactivity coefficients, and power distributions in lattices of slightly enriched annular, uranium dioxide, superheat fuel has been demonstrated by critical experiments on both uniform and clumped arrays of annular superheat fuel[1, 2]. The flow of information between the various calculations is shown on Figure 5.3.

After the fuel rod cell is defined, the following steps are performed to obtain the reactivity and power distributions:

1. The U-238 Doppler broadening, the narrow resonance and infinite mass effective potential scattering, and the Dancoff correction for fuel rod interactions are utilized as input to a Fourier transform fast spectrum calculation. Individual isotopic parameters are weighted

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1. Petersen, G. T., and Warzek, F. W., "AEC Superheat Criticals - A Comparison of Experiment and Theory on Uniform Lattices", GEAP-3882, January 1962.

2. Petersen, G. T., EVESR Preliminary Critical Report, to be published about 8/62 and made available to AEC.

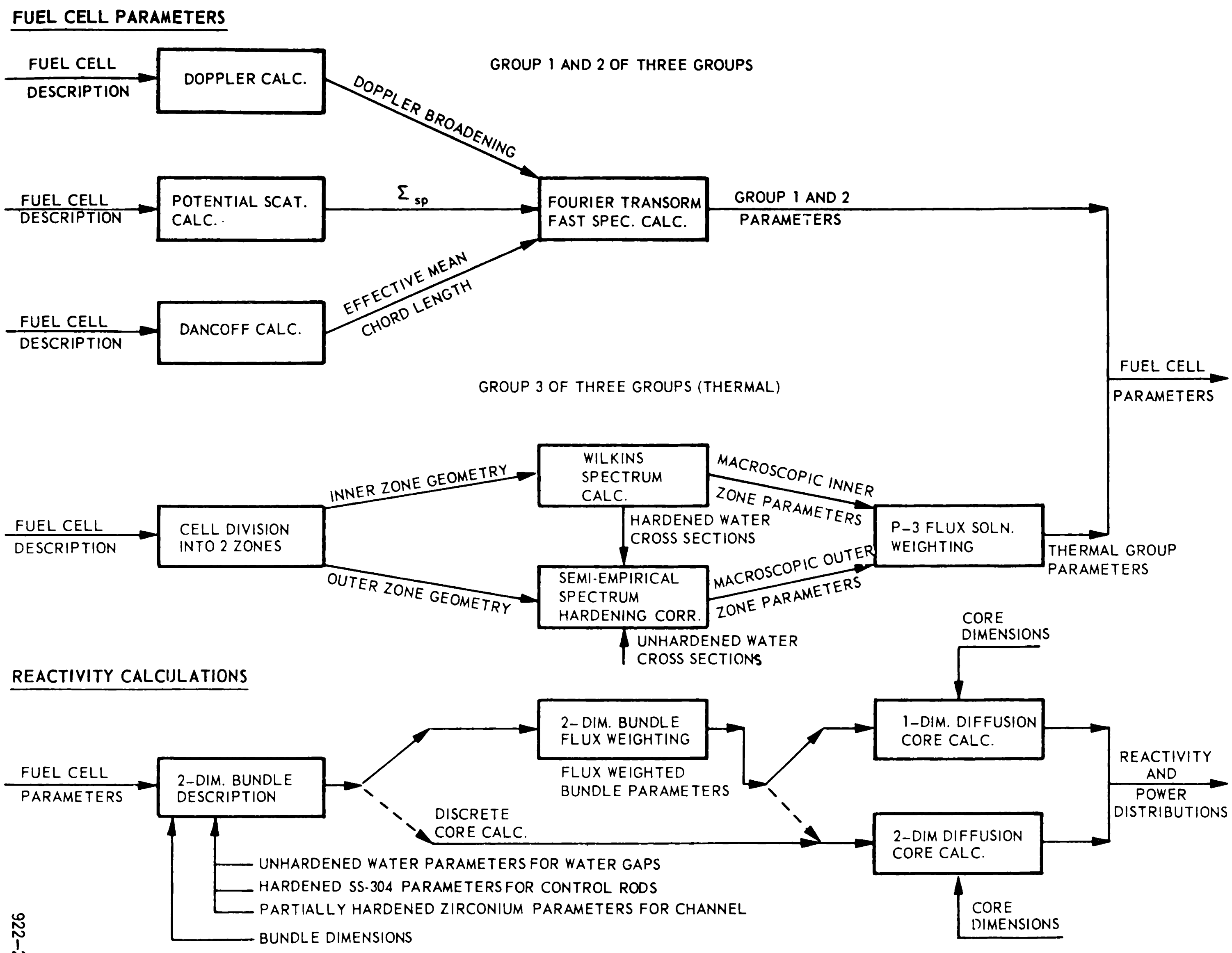

FIGURE 5.3

PHYSICS DESIGN CALCULATIONAL MODEL

over the resulting spectrum and combined to determine the macroscopic parameters for two upper energy groups. The first energy group extends from fission energy to 5.5 kev and the second from 5.5 kev to 0.625 ev (the threshold of the thermal group).

2. Analytic and experimental investigations of the thermal utilization for annular uranium dioxide superheat fuel[1] have shown that to obtain accurate analytic predictions of the thermal utilization when several scattering mean free paths of water are present between fuel surfaces the spatial dependence of the thermal spectrum must be accounted for in the calculation. To include this spatial dependence in the thermal group calculations, a semi-empirical technique which gives good agreement with both thermal utilization measurements and detailed spatially dependent multi-thermal group spectrum calculations was utilized. The basic assumption made is that a continuously variable thermal spectrum within the fuel cell can be approximated by dividing the fuel cell into two zones of constant but dissimilar spectra. The Wilkins heavy gas model is utilized to determine the spectrum within the inner fuel cell zone. The spectrum in the outer fuel cell zone is determined as a function of the outer zone thickness in scattering mean free paths to lie between the hardened inner zone spectrum and a pure water spectrum. Macroscopic spectrum-weighted parameters for both zones are combined and weighted by the P-3 flux solution for the flux depression within the fuel cell to determine the thermal utilization and the thermal group fuel cell parameters.

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1. Petersen, G. T., and Warzek, F. W., "AEC Superheat Criticals - A Comparison of Experiment and Theory on Uniform Lattices", GEAP-3882, January 1962.

3. The presence of structural materials, control rods, or other discontinuities is taken into account by two-dimensional X-Y diffusion theory flux solutions. For reactivity determination, the fuel cell parameters are combined with parameters for other core materials present in a bundle flux weighting calculation. The resulting bundle weighted parameters are used in either one or two-dimensional core-reflector diffusion theory calculations to yield reactivity and gross power distributions. However, for detailed power distribution calculations, the bundle and core-reflector calculations are combined to obtain the full core power distribution with the local interactions included.

5.3 Reactivity Effects of Temperature

The effective multiplication is shown as a function of temperature in both the unflooded and flooded conditions on Figure 5.4. These curves are shown for all control rods withdrawn, for the 12 main control rods fully inserted, and for only the inner four main control rods inserted. These figures apply to the clean core without Xe or Sm.

The temperature coefficients are given in Table 5.3 for several bracketing conditions for the full core. The temperature coefficients experienced by the critical reactor will lie between these bracketing values and are dependent on the exact control rod distribution present. For example, if the control rods are all withdrawn the same distance (ganged pattern) creating a "pancake" critical region with high leakage, the temperature coefficient will be more negative than if alternate control rods are fully withdrawn giving a uniform distribution of controls throughout the core.

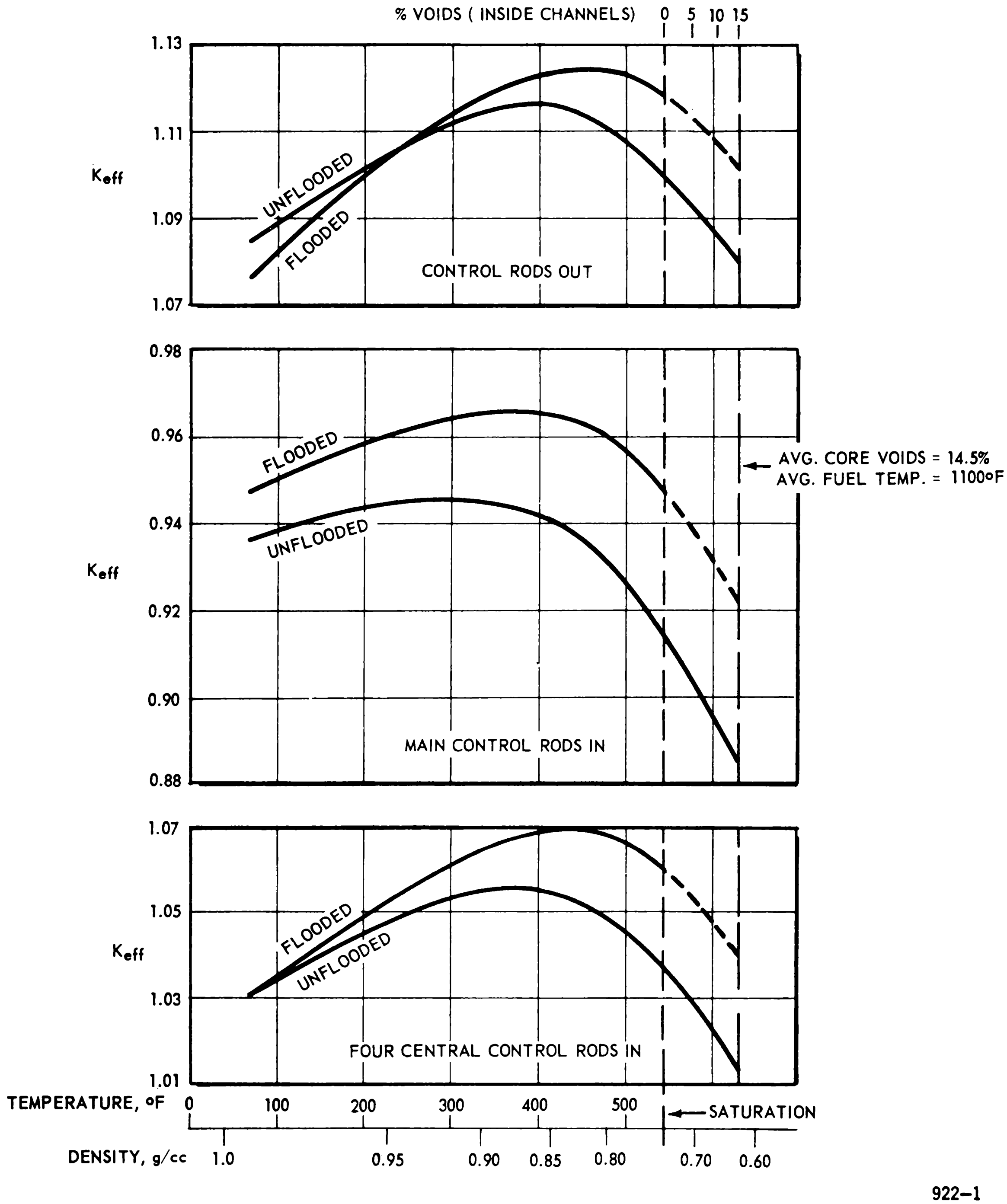


FIGURE 5.4

K_{eff} VERSUS WATER TEMPERATURE

Table 5.3

Moderator Temperature Coefficients
($\Delta k/k$ per $^{\circ}F$ Increase)

	Cold, 68°F		Hot, 545°F	
	Flooded	Unflooded	Flooded	Unflooded
Clean Core Control Rods In	$+1.0 \times 10^{-4}$	$+0.9 \times 10^{-4}$	-2.2×10^{-4}	-2.7×10^{-4}
Clean Core Control Rods Out	$+1.8 \times 10^{-4}$	$+1.6 \times 10^{-4}$	-1.3×10^{-4}	-1.75×10^{-4}

The negative fuel Doppler reactivity coefficient is given for several bracketing conditions in Table 5.4. The fuel temperature dependence of the Doppler coefficient is illustrated on Figure 5.5 for the average hot and cold moderator conditions.

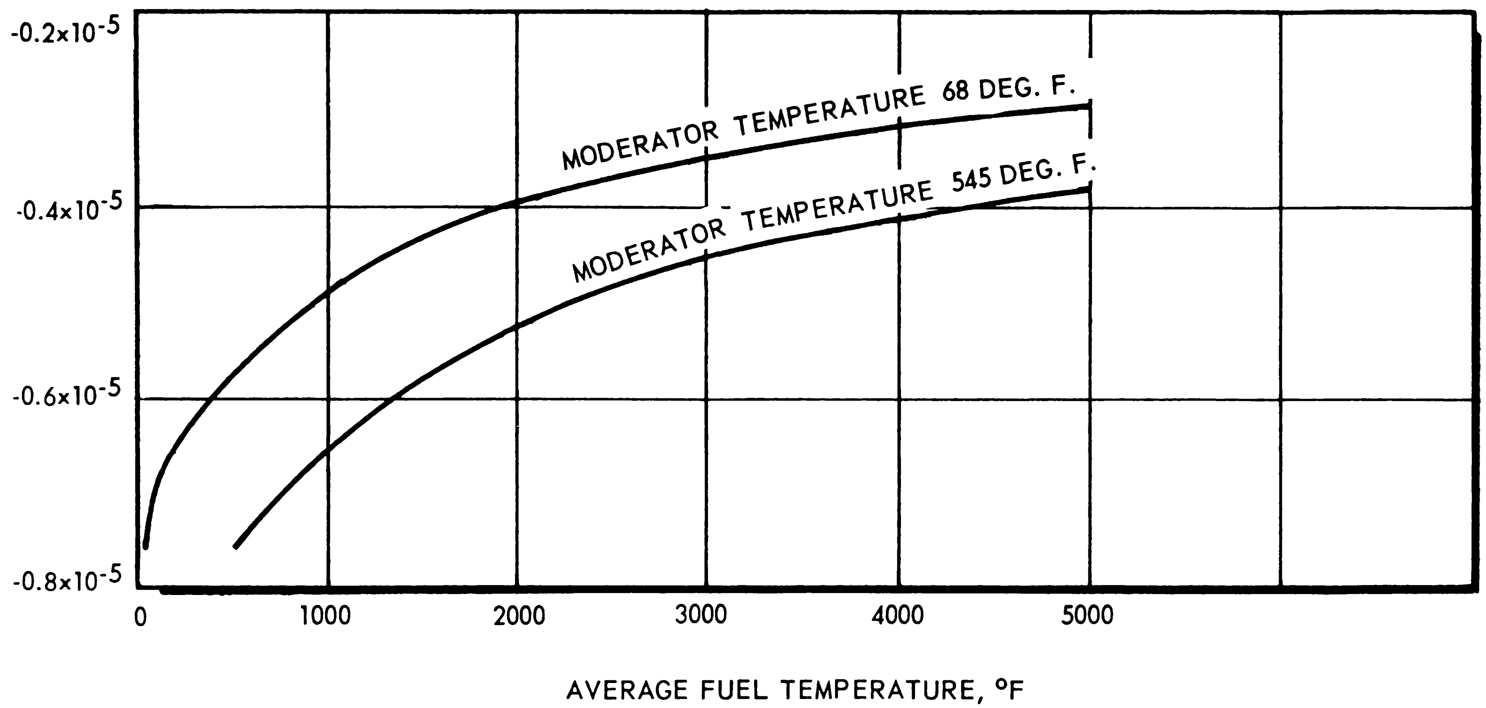
Table 5.4

Doppler Reactivity Effects
($\Delta k/k$ per $^{\circ}F$ Increase)

	Cold, 68°F		Hot, 545°F	
	Flooded	Unflooded	Flooded	Unflooded
Control Rods In	-0.83×10^{-5}	-0.82×10^{-5}	-0.82×10^{-5}	-0.81×10^{-5}
Control Rods Out	-0.72×10^{-5}	-0.70×10^{-5}	-0.72×10^{-5}	-0.69×10^{-5}

5.4 Reactivity Effects of Voids

The moderator void coefficients are given in Table 5.5 for several bracketing reactor conditions for uniform voiding inside the fuel channels (no voiding in the water gaps). The coefficients are given in the units of $\Delta k/k$ per 1 percent void although they were calculated for a void increment of 5 percent void volume. The void coefficients experienced by the critical reactor will lie between these bracketing values and are dependent on the exact control rod distribution present. The values given in Table 5.5 are for the clean core and were calculated including the fuel temperature Doppler effect as well as the voiding reactivity effect.



922-5

FIGURE 5.5
DOPPLER REACTIVITY COEFFICIENT VERSUS FUEL TEMPERATURE

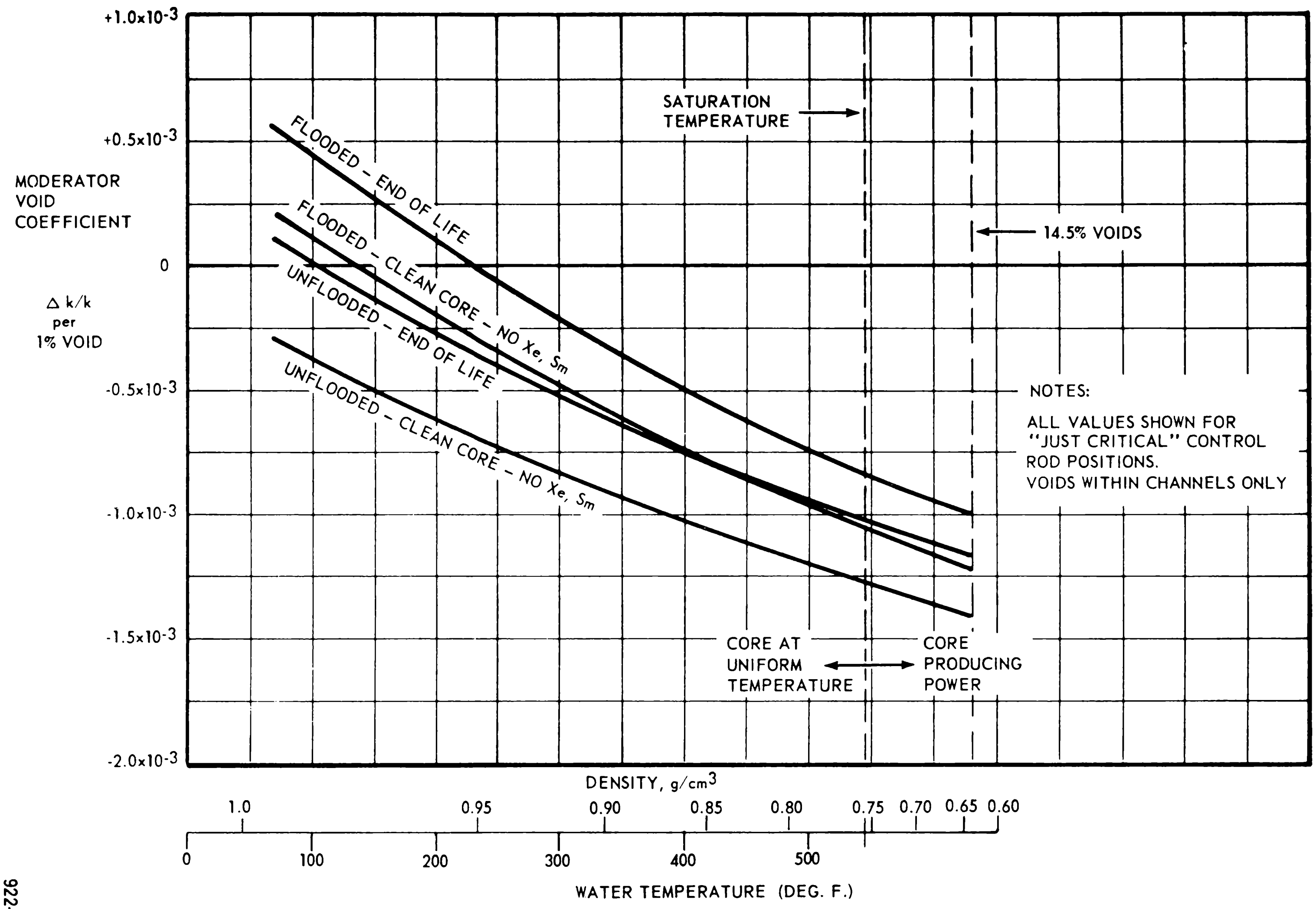
Table 5.5

Moderator Void Coefficient
($\Delta k/k$ per 1% void)

	Cold, 68°F		212°F		Hot, 545°F	
	Flooded	Unflooded	Flooded	Unflooded	Flooded	Unflooded
Control Rods Out	$+0.55\times10^{-3}$	$+0.13\times10^{-3}$	$+0.14\times10^{-3}$	-0.20×10^{-3}	-1.00×10^{-3}	-1.30×10^{-3}
Control Rods In	-0.06×10^{-3}	-0.57×10^{-3}	-0.41×10^{-3}	-0.87×10^{-3}	-1.48×10^{-3}	-1.76×10^{-3}

Figure 5.6 gives the void coefficient as a function of temperature for both the beginning and end of life for the normal control rod withdrawal pattern for the "just critical" unflooded and flooded reactor cores. The normal control rod withdrawal pattern consists of withdrawing the outer 8 main control rods in a banked manner until they are fully withdrawn (with the inner 4 main control rods fully inserted) and the withdrawing the inner 4 main control rods in a banked pattern (with the outer 8 main control rods fully withdrawn). It can be seen that, although a positive void coefficient exists for the flooded condition at room temperature, this coefficient becomes negative prior to atmospheric boiling (212°F), even at the end of life with almost all control rods fully withdrawn. The void coefficients shown in Figure 5.6 include the fuel temperature Doppler effect as well as the voiding reactivity effect. The Doppler effect is very small, however, until the reactor is brought to power to heat the fuel significantly above the moderator temperature.

Since the void coefficients are spatially dependent throughout the reactor, it is possible by selective void formation in the regions of maximum positive void coefficients to achieve a larger positive void effect than



922-34

FIGURE 5.6

MODERATOR VOID COEFFICIENT VERSUS MODERATOR TEMPERATURE

the average coefficients would indicate. For reference, the voiding reactivity effect as a function of void volume percent, with all process tubes flooded at 212°F, and all control rods withdrawn, is given as Case A on Figure 5.7. The flooded, 212°F condition was chosen since this is the condition where the potential for adding void reactivity in a nuclear excursion is the greatest. Voids are homogeneously distributed within the fuel channels (no voids in the water gaps). It can be seen that the maximum reactivity increase encountered in this condition by this void distribution is +0.2 percent Δk. If, however, the voids were formed predominantly in the center of the core the voiding effect is slightly more positive. Case B of Figure 5.7 shows the voiding reactivity effect as a function of the void volume percent with all process tubes flooded at 212°F and all control rods withdrawn, for void formation only in the central quarter of the core. The voids in this core region were homogeneously distributed within the fuel channels (no voids in the water gaps). The maximum reactivity increase encountered in this condition by void formation is +0.4 percent Δk compared to +0.2 percent Δk for the voids distributed over the entire core. Insertion of control rods, however, will significantly lower the positive void reactivity effect. Case C of Figure 5.7 shows the voiding reactivity effect as a function of the void volume percent with all process tubes flooded at 212°F, with the outer 8 main control rods inserted and the inner 4 main control rods withdrawn, for void formation only in the central quarter of the core. Again the voids in the region were homogeneously distributed within the fuel channels (no voids in the water gaps). The maximum reactivity increase encountered in this condition by void formation is +0.1 percent Δk compared to +0.4 for the same reactor

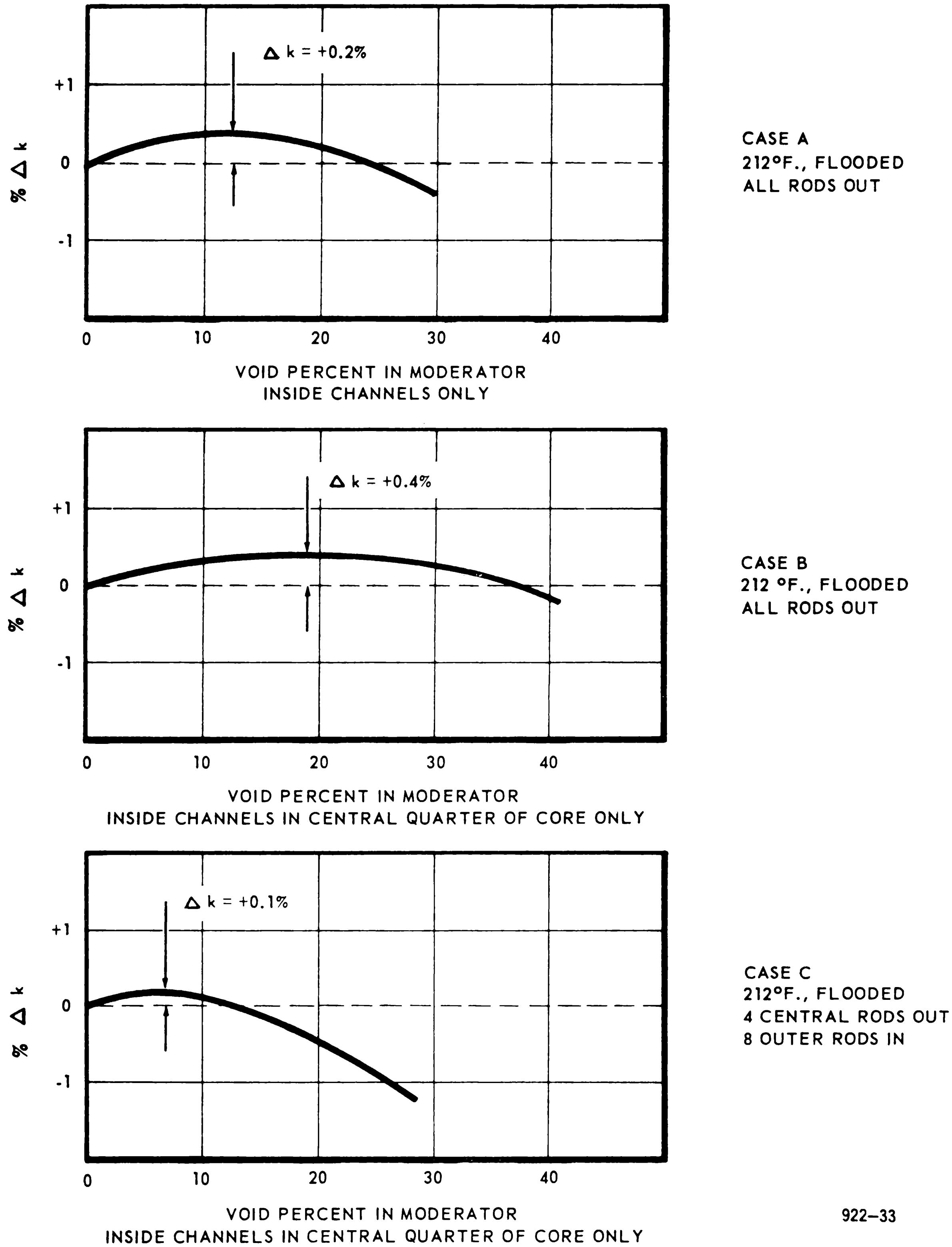


FIGURE 5.7
VOIDING REACTIVITY EFFECT VERSUS PERCENT VOIDING WITHIN CHANNELS
(No Voiding in Water Gaps)

condition with all control rods withdrawn. In no case does the net reactivity increase approach one dollar.

If, however, voids are formed in the water gaps as well as within the fuel channels, a slightly larger positive void reactivity effect can be attained. Significant void formation within the water gaps, as well as within the fuel channels, can only occur from direct heating of the moderator water by prompt neutron and gamma energy in a nuclear excursion or as the result of a rapid pressure decrease. (Even in a nuclear excursion, heat conduction from the fuel to process tubes to water within the channel quickly shifts the void formation ratio heavily back toward the normal operating situation where voids are formed only within the channels.) Only in the nuclear excursion could this voiding also be preferentially located within the center of the core. However, in a nuclear excursion severe enough to cause a significant uniform enthalpy rise in the water prior to significant heat conduction, void formation is suppressed by a local pressure rise due to the thermal expansion and resulting acceleration of the water. In fact, it is believed that the formation of voids would occur only after the exponential power increase is terminated in most instances. Nevertheless, it is important to place upper limits on the positive void effect that could occur if voids were formed uniformly in the water gaps as well as within the channels preferentially toward the center of the core. Case A of Figure 5.8 shows the effect of uniform void formation in the water gaps and inside the channels over the entire core. The maximum positive void reactivity effect is +0.7 percent Δk compared to +0.2 percent Δk for the same case with void formation only inside the channels. Case B of Figure 5.8 shows the effect of uniform void formation in the water gaps

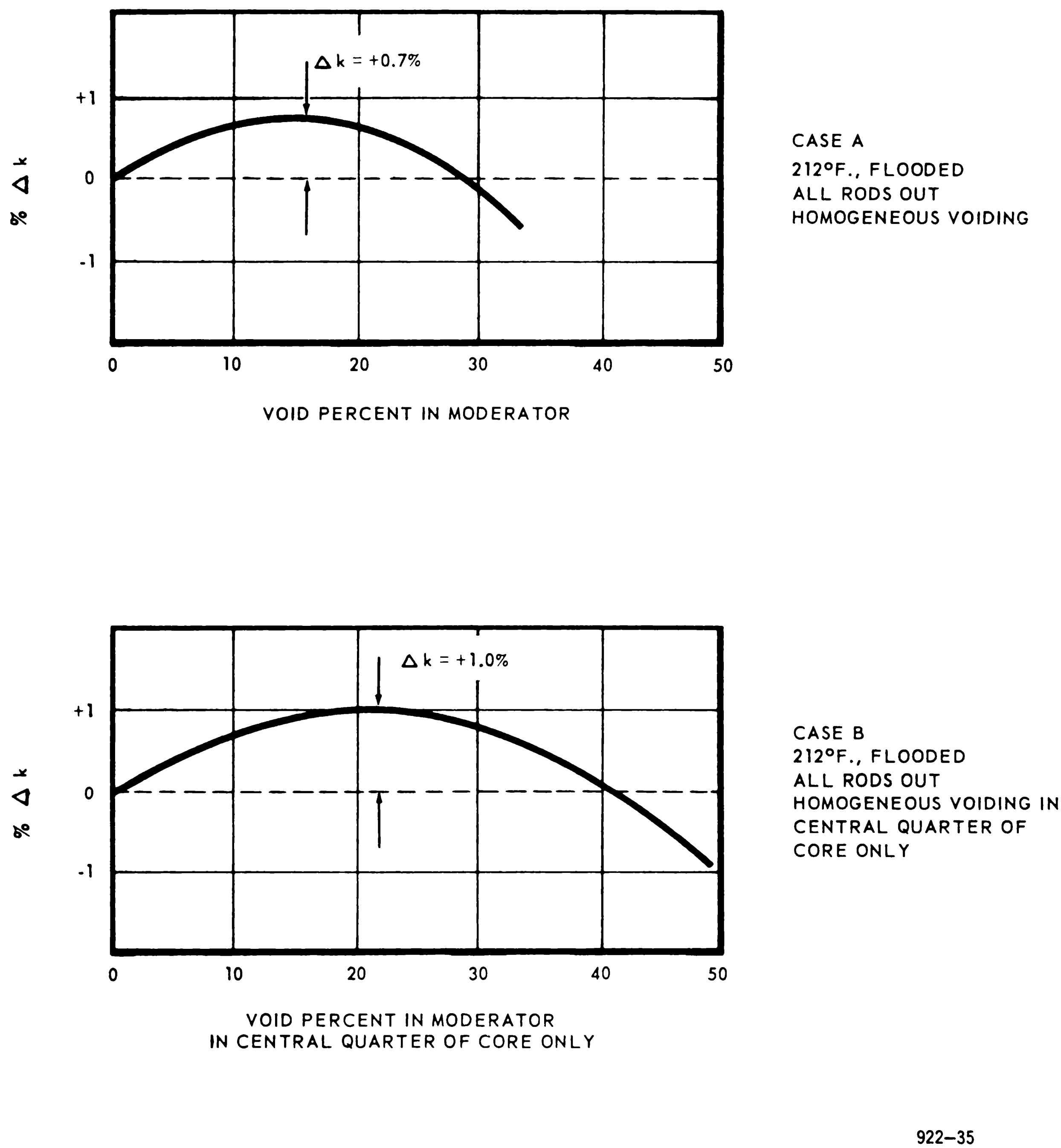


FIGURE 5.8

VOIDING REACTIVITY EFFECT VERSUS PERCENT VOIDS IN ALL CORE WATER

(Including Water Gaps)

and inside the channels but only in the central quarter of the core. The maximum positive void effect is +1.0 percent Δk compared to +0.7 percent Δk for uniform void formation over the entire core. The presence of control rods in the core would significantly lower these positive effects. It is quite significant that if unflooding should accompany the void formation the net reactivity effect would not increase. This is because at 212°F the unflooding effect itself is near zero or negative, and becomes more negative when moderator voids are formed. For example, calculations indicate that unflooding the full core at 212°F, with all control rods withdrawn and 11 percent voids in all the water, results in a negative reactivity change of -0.4 percent Δk. Therefore, the maximum conceivable positive reactivity addition due to the combined effects of void formation and unflooding at 212°F or higher is not expected to exceed +1.0 percent Δk.

5.5 Reactivity Effects of Flooding

As indicated above, the possibility of adding reactivity from flooding or unflooding the steam passages of the core was a paramount consideration in the nuclear design. Since reactivity could be added by either flooding or unflooding, depending on moderator temperature and control rod positions, it was important to optimize the nuclear design so that these mechanisms for reactivity addition would not significantly detract from the overall plant safety. The optimization considered not only the total achievable reactivity addition, but other accident considerations as well, which resulted in a nuclear design which minimizes the total unflooding effect to a greater extent than the total flooding effect.

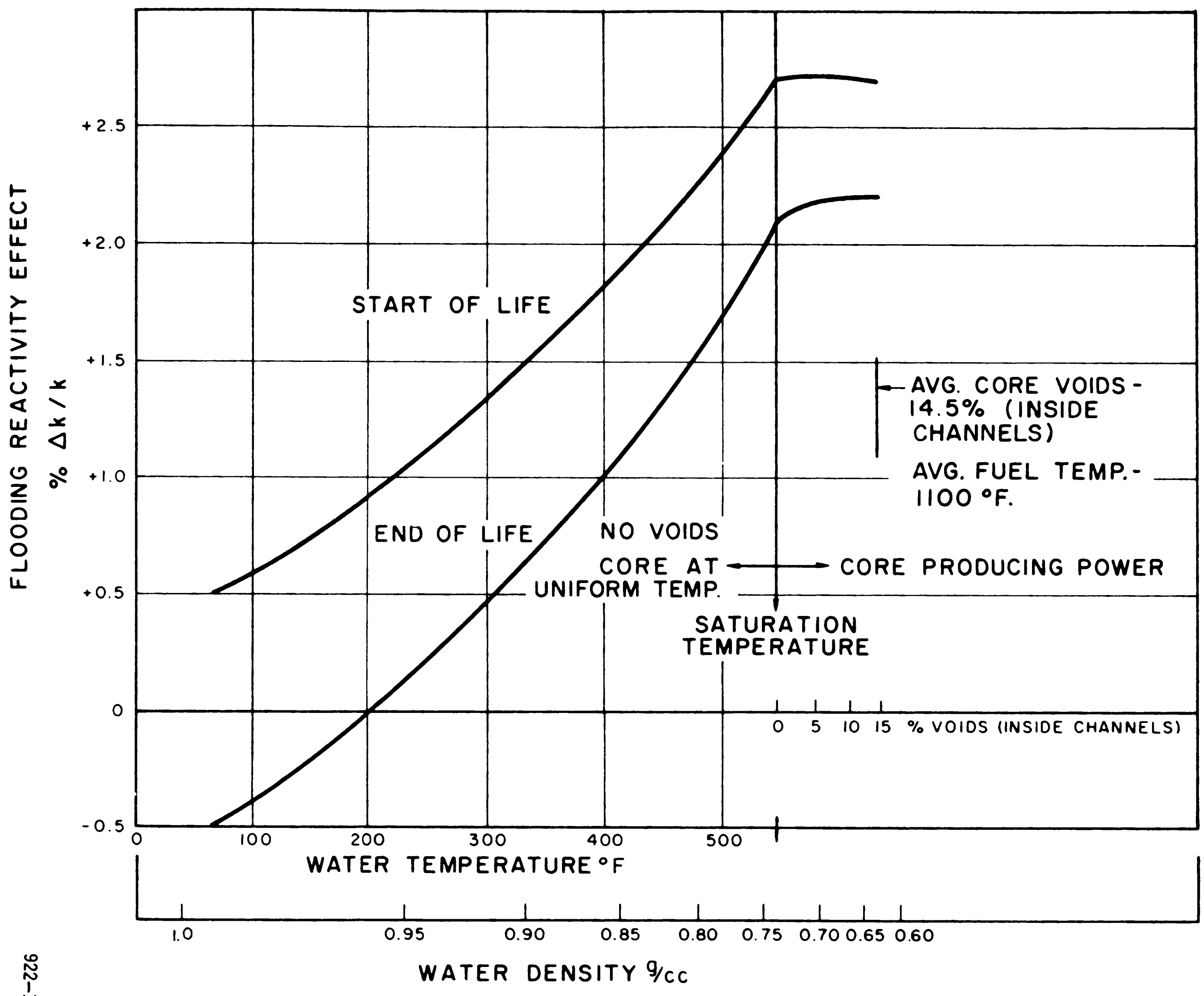
Figure 5.9 shows the flooding reactivity effect as a function of moderator temperature at the start of life and end of life. Flooding reactivity is shown for a core that is just critical in the unflooded condition for normal control rod patterns. "Normal control rod patterns" assume the eight outer main control rods are withdrawn in a banked fashion until criticality is achieved. As xenon, samarium, and fission products build up, the eight outer control rods are fully withdrawn and the four inner main control rods are then withdrawn in a banked manner until criticality is achieved. For example, consider the start-of-life cold flooding effect. A normal control pattern with the four central rods fully inserted, and the eight outer rods approximately 40 percent inserted, is expected to result in a just-critical core at 68°F, unflooded. This could be considered a realistic core condition during some of the startup critical tests. For this control rod pattern, the reactivity effect of flooding the superheat passages is +0.5 percent Δk/k. Table 5.6 illustrates the maximum unflooding or flooding effects which may be achieved for the Mark II core.

Table 5.6

Maximum Unflooding and Flooding Reactivity Effects

	Control Rod Positions		Reactivity Effect
	4 Central Rods	8 Outer Rods	% Δk/k
Unflooding, cold, end-of-life	Out	Out	+0.7
Flooding, hot, start-of-life	In	40% In	+2.7
Flooding, hot, start-of-life	80% In*	80% In*	+3.1

* Uniform banking of control rods is a pattern which is procedurally avoided during normal operation of the reactor.



922-16

FIGURE 5.9
FLOODING REACTIVITY EFFECT VERSUS WATER TEMPERATURE
(XENON-FREE CORE)

Perturbation calculations were made to predict the differential reactivity effect as a function of core radius for unflooding the cold core and flooding the hot core. These calculations show that for both of these cases the reactivity effect per unit of core volume is greatest in the center of the core. Table 5.7 illustrates the results of these calculations in terms of the percent of the available total flooding or unflooding reactivity which is added upon flooding or unflooding various portions of the core, respectively.

Table 5.7

Effect of Core Location on Flooding-Unflooding Reactivity
(All Control Rods Out)

Core Location	Percent of Total Cold Unflooding Reactivity Effect	Percent of Total Hot Flooding Reactivity Effect
Center 30%	40%	50%
Middle 30%	30%	30%
Outer 40%	30%	20%

5.6 Reactivity and Control Requirements

The EVESR initial core is enriched to yield an average core lifetime of 4000 megawatt days per ton. Figure 4.8 illustrates the expected enrichment by rod. The average enrichment is approximately 5.4 percent U-235 by weight. The tabulation below illustrates the reactivity requirements to provide 4000 MWD/T exposure and 12.5 MW(t) power operation at the end of core life.

4000 MWD/T Burnup	0.058 Δk
Equilibrium Xe	0.015 Δk
Equilibrium Sm	0.007 Δk
Total Lifetime Effects (Required initial excess k at hot clean full power condition)	0.080 Δk
$450^{\circ}F$ Flooded to $545^{\circ}F$ Unflooded	0.024 Δk
14.5% Voids	0.016 Δk
Fuel Doppler $545^{\circ}F$ to $1250^{\circ}F$	0.005 Δk
Total Operating Effects (Reactivity defects from peak k_{eff} to full power for temperature, voids, unflooding and Doppler)	0.045 Δk
Total Reactivity Requirements (See Figure 5.4 also)	0.125 Δk

6.0 SUB-TASK C-1 FUEL PROCESS DEVELOPMENT AND FABRICATION

6.1 Status Report

The division of responsibility for the fabrication of the Mark II core load has been established, a fabrication schedule has been outlined, and tooling has been initiated. Procurement of cladding materials, riser-downcomers, reducers, upper tube sheet, lower tube sheet support frame, etc., has been started.

7.0 SUB-TASK C-2 FUEL ELEMENT PROCESS DEVELOPMENT AND FABRICATION

7.1 Material Procurement

On July 27, 1962, a meeting was held to review bids which had been received for the EVESR cladding and end plug materials. With the information available, it was decided that Carpenter Steel Company, located at Reading, Pennsylvania, submitted the most suitable bid. Efforts began immediately to secure approval for placing of the order.

Carpenter Steel promised to complete delivery of all materials within 20 weeks after receipt of an order, with partial deliveries starting in 9 to 11 weeks. They have taken no exceptions to APED material specifications.

Two sources have been developed for the procurement of vacuum melted (low nitrogen) Type 304 and Type 310 stainless steel tubing in sizes required for testing as CL-1 heater sheaths and as prototype fuel elements. Nuclear Metals, Inc., and Carpenter Steel Company will each supply tubing with a geometry of 0.565 inch - 0.570 inch O.D. x 0.501 inch $\pm$ 0.001 inch I.D. x 42 inches long and with the following chemical limits:

1. Type 304 S.S.

   N = 0.010 percent max.
   C = 0.02 percent max.

2. Type 304 S.S.

   N = 0.010 percent max.
   C = 0.08 percent max.

3. Type 310 S.S.

   N = 0.010 percent max.
   C = 0.02 percent max.

4. Type 310 S.S.

N = 0.010 percent max.
C = 0.25 percent max.

This tubing should be delivered by August 31, 1962. Four ingots of the high purity 304 and 310 will be melted by General Electric Research Laboratory and supplied to Nuclear Metals for tube drawing.

In addition, an attempt was made to establish a source of vacuum melted material on the West Coast. Wah Chang Corporation's plant in Albany, Oregon was contacted and their facilities inspected. They have complete facilities for vacuum melting both by the electron beam and the consumable electrode methods and can roll strip to appropriate sizes for tube forming. However, Wah Chang is a producer of refractor metals, and although they are interested in our vacuum melted material, their personnel have had little experience with stainless steels. For this reason, they must be considered as a future source, if required, rather than an immediate source.

7.2 Fabrication Development

A specification for pickling, cleaning, and passivating of the EVESR fuel segments has been completed and circulated for comment. The final specification will be ready by August 31, 1962.

Welding development for the EVESR segments has included all of the alloys under consideration. Final parameters have been determined for the Type 304 S.S. welds and work is continuing to finalize the welding procedure for Inconel, Incoloy, Type 310 S.S. and Type 347 S.S. Preliminary Process Instructions for all end plug welding should be completed by August 31, 1962.

7.3 EVESR Fuel Element and Design Assistance

7.3.1 Outer Clad Swaging

Three 12-inch long annular specimens (1-1/4 inch O.D.) have been swaged with little or no pellet damage. In all three cases, pellet chips were disentegrated to fine powder by the swaging action. Maximum depth of chipping was approximately 1/32 inch; efforts are being made to minimize chipping. Experimental parameters were as follows:

Clad Material	Average Diametral Loading Gapm	Average Diametral Swaged Gap	Average Swaging Reduction
304	.0125	.0045	.0075**
304	.0135	.0045	.0085*

* Two swaging reductions used.
** One swaging reduction used.

Maximum variation in the O.D. size along the entire tube was .001 inch. Bow (one-half "runout" Measured with a dial indicator while rotating the element on V-blocks at either end) was .005 inch maximum.

Evaluation of pellet condition of the first specimen (after swaging) was obscured by damage during removal of the outer cladding. New tooling has been developed to permit clad removal and ascertain the condition of the swaged over pellets.

7.3.2 Inner Clad Expansion

The technique for reducing the inner clad-to-pellet gap by pressurizing the completed element is feasible. An average diametral gap of between .001 inch and .002 inch can be achieved by pressurizing in the range of 3300 to 3800 psi (at room temperature with all materials in the annealed condition). Experiments indicate that ovality of the expanded clad should not exceed .001 inch. Three 12-inch capsules (304 clad) have been pressurized and then been inspected for pellet damage. Results are as follows:

Specimen	Pressure	Inner Dia. Gap	Pellet Condition
No. 2	3200	.003	None Broken
No. 3	3200	.0025	None Broken
No. 4	3300	.0018	None Broken

Based on the nominal EVESR pellet size (.719 ± .001) and the inner clad size (.714 ± .0025) the maximum possible as-loaded gap should be about .008 inch. The largest possible expansion should then be .006 inch to .007 inch. Tests have been made to approximate the residual stresses in a .750 inch O.D. 304 clad after a .006 inch expansion. The maximum residual tensile stress on the clad inner fiber was calculated to be about 1250 psi. Hardness tests on original and expanded tubing showed an average increase of about two of the Rockwell "B" scale (73 to 75).

8.0 SUB-TASK C-3 NEUTRON SOURCE DESIGN AND FABRICATION

8.1 Status Report

Drawings, specifications, and shop orders for the fabrication of the neutron source have been issued. The design of the source holder will be initiated soon.

9.0 SUB-TASK C-4 FUEL EVALUATION INSTRUMENTATION

9.1 Instrumentation for Initial Core

The initial scoping study of the EVESR first core instrumentation has been essentially completed. This instrumentation consists of the following:

No. Fuel Assemblies	Type of Instrumentation
8	Mid-pass and exit steam temperature for all elements in fuel assembly.
5	Ten flux wire monitor tubes attached to the outer surface of the process tubes in each fuel assembly.
3	Fuel assembly moderator inlet temperature.
3	Fuel assembly steam inlet temperature at the inlet plenum and steam exit temperature at the outlet steam plenum.

The above instrumentation is concentrated in eight fuel assemblies located in a quarter core arrangement. In addition, there will be a special highly instrumented fuel assembly which will contain all of the above measurements plus six clad thermocouples and three flow measurements from the three typical elements in an assembly of nine. One assembly in the core will be equipped with two vibrometers located on the riser-downcomer in the two principal planes of vibration. The remaining fuel assemblies in the core are monitored from outside the pressure vessel. Since the steam outlet from each fuel assembly exits separately from the pressure vessel, it is possible to measure and control each individual assembly flow rate and exit steam temperature from a point external to the pressure vessel. These three measurements plus moderator feed flow rate, moderator inlet temperature to pressure vessel and moderator inlet temperature to the fuel assemblies permits a crude analytical evaluation of the fuel assembly and individual fuel element performance to be made. The extensive in-core instrumentation provides a more accurate check on the actual fuel performance in the core.

The mode of reactor operation for the cladding evaluation tests will be as follows:

1. The main console operator will hold a constant reactor power level on EVESR using the control rods and ion chamber indicator readings.

2. The R&D console operator will monitor in the control room all the individual fuel assembly flow rates and exit temperatures and will have control over the fuel assembly flow rates through the use of twelve bias flow control valves. His console will consist of four fast four-point recorders (one point each for uncorrected flow rate, density compensated flow rate, exit pressure, and exit temperature), a steam header selector switch and a single valve controller. As many as four fuel assemblies are tied to a single exit header. A bias flow control valve in the header controls the flow in all four fuel assemblies simultaneously. There are twelve such headers and bias flow control valves except that in certain fuel assembly locations less than four assemblies feed into the header.

   The steam header selector switch mentioned above has twelve positions corresponding to these twelve bias flow control valves and headers. Any of the twelve bias valves can be dialed directly and positioned at will using the single valve controller mentioned earlier. At the same time that the valve is dialed in, all the fuel assemblies influenced by that particular bias valve will be switched to the four fast four-point recorders for direct surveillance of changes in flow rate, pressure and temperature. The R&D console operator will use this system to maintain certain predetermined peak clad temperatures for each bundle.

In addition to this control and surveillance instrumentation, the R&D console will also record continuously the flow rate, exit temperature and pressure from all thirty-two fuel assemblies. All of the in-core instrument readings installed in the fuel assemblies will also be recorded on the R&D console.

Procurement of thermocouples and thermowells for installation in the fuel assemblies has been initiated.

10.0 SUB-TASK C-5 FUEL TESTS - CHEMISTRY SAMPLING STATIONS DESIGN AND INSTALLATION

10.1 Status Report

A preliminary rough draft scoping document has been issued for comment covering the proposed sampling and test equipment and the proposed types of VAL-Chemistry work effort. This document is presently under review and will be reported in detail when the scope of work has been agreed upon and accepted.

11.0 SUB-TASK D-1 FUEL TESTS - ENGINEERING PHYSICS

11.1 Status Report

The initial loading schedule and critical test program has been outlined for the initial core loading. Detailed procedure and test planning remain to be prepared in accordance with this outline.

12.0 SUB-TASK D-2 FUEL TEST - SITE OPERATIONS

12.1 Status Report

No work will be carried out in this work area until the start of the fuel loading activity.

13.0 SUB-TASK D-3 FUEL TESTS - TEST PROCEDURES

13.1 Status Report

Considerable engineering effort has been devoted in this area in connection with the EVESR planning effort and the EVESR safeguards effort, but the actual preparation of the detailed test procedures for the program is not scheduled to start until later.

14.0 SUB-TASK D-4 FUEL TESTS - DATA REDUCTION

14.1 Status Report

Work has been initiated to expand the present TAPS code, which analyzes a single fuel element for heat fluxes and temperature distributions, to cover a bundle of nine fuel elements. The ultimate objective will be to develop both the physics and heat transfer codes such that raw test data can be keypunched and used directly in the machine calculations of fuel performance in EVESR. This type of automation is necessary for handling the large masses of data that will be generated in EVESR. This program will also be used to help set the desired operating limits for the fuel test program.

15.0 SUB-TASK D-5 FUEL ACTIVITY RELEASE AND COOLANT CHEMISTRY

15.1 Status Report

A preliminary rough draft scoping document has been issued for comment covering the proposed sampling and test equipment and the proposed types of VAL-Chemistry work effort. This document is presently under review and will be reported in detail when the scope of work has been agreed upon and accepted.

16.0 SUB-TASK E-1 POST-IRRADIATION EXAMINATION PLANNING AND EVALUATION

16.1 Status Report

The details of the type of post-irradiation examinations of failed fuel elements which are to be carried out will follow after the nature of each individual fuel failure is better defined. To aid in the study of fuel failures as they do occur, all the fuel will be given a pre-irradiation examination during fabrication. Detailed measurements of the outside diameter and the inside diameter at a number of locations will be taken as well as length and bow measurements. The first ten elements to be measured by manufacturing personnel will be measured concurrently with RML personnel in order to coordinate and calibrate the measurements for comparison with post-irradiation measurements. During manufacture of the fuel elements and assembly into fuel bundles, appropriate orientation marks will be used to orient the elements in the process tubes and to orient the bundles in the core. A record will be kept of those elements containing corrosion coupons in the velocity booster tube.

17.0 SUB-TASK E-2 PRE- and POST-IRRADIATION EXAMINATION

17.1 Status Report

Except for the pre-irradiation examination discussed under Task E-1, no effort in this area will occur until after the first EVESR fuel failure occurs.

18.0 SUB-TASK F ADVANCED FUEL DESIGN AND FABRICATION

18.1 Status Report

It is presently anticipated that during the course of the contract program, twenty advanced fuel assemblies will be designed, manufactured and irradiated. The design of these twenty assemblies shall be agreed upon by AEC and General Electric Company, with the twin objectives of further defining the performance limits of superheat fuel and of developing a superheat fuel which is economically attractive for use in large central stations.

APPENDIX I

TASK REFERENCE SHEETS AND PROGRAM SCHEDULES

Sub-task A-1: Program Planning and Evaluation

A. Objective: To provide the necessary planning and technical direction on a continuing basis for the Superheat Fuel Development Program to assure that all development work has a specific direction and objectives that are realistic and of direct significance to the early achievement of low cost power from nuclear superheat reactors. This task applies to both initial and advanced EVESR fuel.

B. Approach to Problem:

1. Make the necessary studies and evaluations to establish detailed program objectives including the desired fuel performance levels, define and assign specific tasks, and establish schedules and budgets for each phase of the EVESR Superheat Fuel Development Program.

2. Evaluate development results from each contributing group as well as from other related AEC activities and assure that new data is being utilized as it becomes available and that any desirable program changes are made to achieve the objectives on schedule and within budget.

3. Determine and recommend to the AEC the specific fuel designs to be tested. Establish the objectives, test conditions and schedules for all EVESR tests.

4. Evaluate results, review, edit and issue reports.

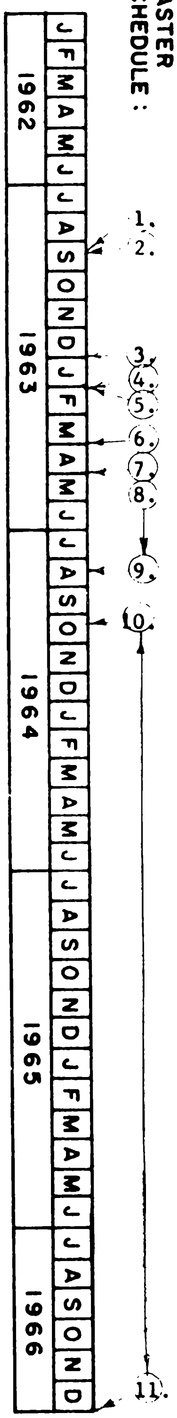
C. Expected Results: A vigorous, flexible and well directed EVESR Superheat Fuel Development effort which will yield the necessary technology to achieve low cost power from nuclear superheat reactors.

TASK A (Sheet #1)
Program Planning & Evaluation
Reference Information on Significant
Mileposts for EVESR Construction

EVESR R & D PROGRAM

MASTER SCHEDULE :

ACTUAL :



1. Transmittal of Safeguards Report to ACRS
2. Initiation of Fuel Fabrication
3. Completion of On-Site Construction
4. Receipt of Operating License
5. Completion of Initial Fuel Loading
6. Initiation of Advanced Fuel Fabrication
7. Completion-Start-Up Tests
8. Initiation Power Operation-Reactor Evaluation Tests
9. Initiation of EVESR-VBWR Joint Operation
10. First Loading of Adv. Fuel

Continue Advanced Fuel & Clad Evaluation Program
Irradiate & Examine Failed Elements
Study System Contamination

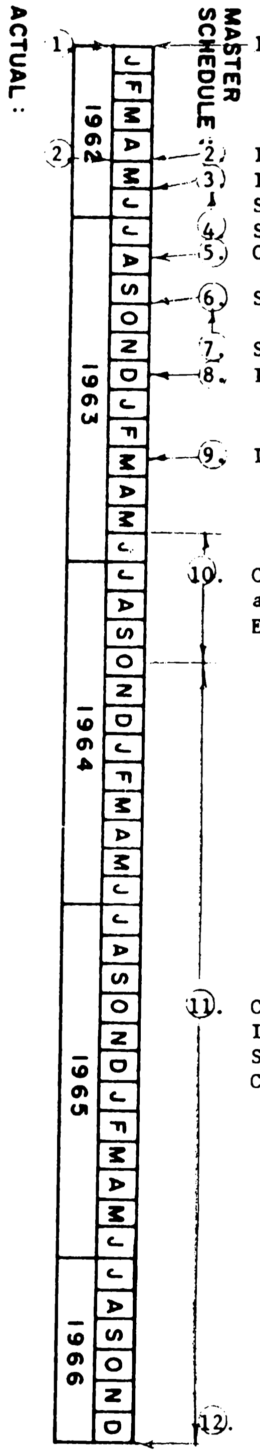
11. End of Program

DATE : August 31, 1962
REV. 0 DATED: Aug. 31,'62

TASK A (Sheet #2)

PROGRAM PLANNING & EVALUATION

EVESR R & D PROGRAM



1. Initiate Planning, Budgeting
   Start Mark II Fuel Design
2. Issue Program Scope
3. Initiate Chemistry & Instrumentation Scoping Studies
4. Start Data Reduction Studies
5. Complete Scope Chem. & Instruments
6. Start Advanced Fuel Design
7. Start VBWR Tie-On Study
8. Issue Preliminary Start-Up Test Proc.
9. Issue R & D Test Procedures
10. Complete Startup Tests and Reactor Evaluation Experiments.
11. Continue Advanced Fuel Program
    Irradiate & Examine Failed Elements
    Study System Contaminations
    Complete Fuel Cladding Eval.
12. End of Program

DATE : August 31, 1962

REV. 0 DATED: Aug. 31,1962

Sub-task B-1: Fuel Design

A. Objective: To provide the detailed mechanical, thermal and hydraulic design for the EVESR initial fuel and spare fuel bundles, as well as the necessary fuel accident analyses and safeguards evaluations.

B. Approach to Problem:

1. Evaluate the results of cladding material screening evaluations, fuel irradiation tests, heat transfer and hydraulic tests, fabrication development, physics and other development work. Incorporate the various development results into a fuel bundle design which has the best potential for satisfactory performance in nuclear superheat environment.

2. Conduct the necessary design studies, engineering development and heat transfer tests, and produce shop drawings and specifications for the EVESR initial fuel, including instrumentation. Work with NEPS, on instrument designs that effect fuel or reactor designs.

3. Perform the necessary fuel accident analysis and safeguards evaluations.

4. Establish fuel operating limits to assure the safe operation of the reactor while performing the fuel development tests.

C. Expected Results: This task should yield complete design and performance data which will permit fabrication, installation, licensing and safe EVESR operation with the initial core.

Sub-task B-2: Fuel Design, Engineering Physics

A. Objective: To provide the necessary engineering physics effort to complete the design, accident analysis, and safeguards evaluation and fabrication of the initial EVESR fuel (Mark II fuel).

B. Approach to Problem:

1. Provide nuclear design parameters for incorporation and co-ordination with the mechanical and thermodynamic design of the fuel.

2. Prepare nuclear specifications for the fabrication of fuel and check "as-built" fuel for compliance with specifications (shop following).

3. Evaluate all fuel designs for potentially hazardous operating conditions.

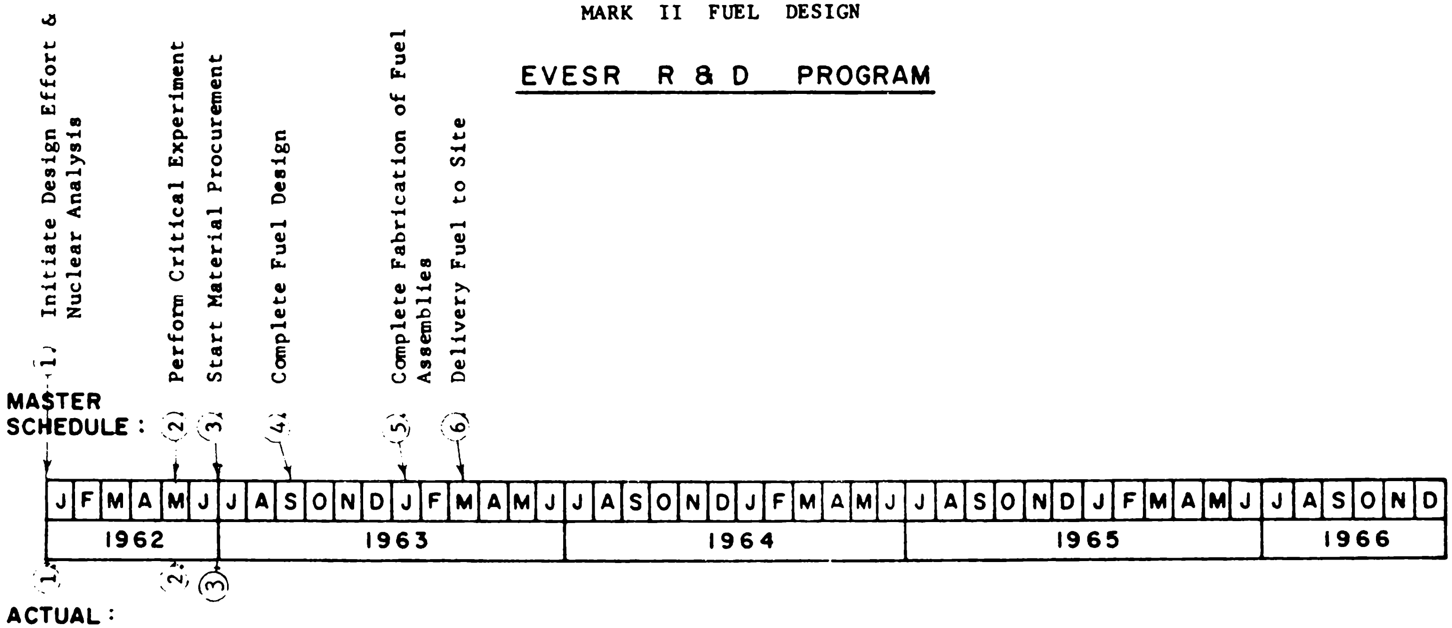
4. Establish the operating parameters and limitations for the EVESR fuel.

C. Expected Results: A satisfactory EVESR fuel design and safeguards evaluation from a physics standpoint.

TASK B

MARK II FUEL DESIGN

EVESR R & D PROGRAM



DATE : August 31, 1962

REV. 0 DATED: Aug. 31, 1962

Sub-task C-1: Fuel Process Development and Fabrication

A. Objective: To provide overall control and responsibility for the process development and fabrication of the initial EVESR fuel.

B. Approach to Problem:

1. Direct and co-ordinate the EVESR fuel process development and fabrication tasks which are assigned as follows:

   a. Fuel pellet process development, materials procurement, and pellet fabrication: Fuel Manufacturing Operation (FMO).

   b. Fuel element (rod) process development, clad and end plug procurement, and rod fabrication: Fuels and Materials Development - VAL.

   c. Fuel bundle process development, materials procurement, and bundle fabrication and assembly: Fuel Manufacturing Operation.

2. Provide all necessary specifications, drawings and engineering instructions for the above work.

3. Approve the specific areas of process development by FMO and Fuels and Materials Development, and follow development work.

4. Follow progress of fuel fabrication through purchasing and shops to assure compliance with drawings and specifications, including installation of fuel evaluation instrumentation.

C. Expected Results: Delivery of satisfactory EVESR fuel on schedule and within budgeted costs.

Sub-task C-2: Fuel Element Process Development and Fabrication

A. Objective: To deliver fuel elements (rods) meeting the design and specifications provided by the Engineering Development Subsection for the initial EVESR fuel.

This task, C-2, consists of procuring fuel clad and end plug materials, performing the necessary process development to establish the fuel element (rod) fabrication process, and fabricating the fuel elements for the initial EVESR loading. Fuel pellet materials procurements, process development and fabrication will be performed by Fuel Manufacturing Operation (FMO), and are not included in this Task C-2. FMO will also fabricate and assemble the fuel bundles.

B. Approach to Problem:

1. Procure the necessary cladding and end plug materials in accordance with product specifications furnished by the Engineering Development Subsection. Coordinate the procurement of materials for these components with the procurement of materials for matching components by Fuel Manufacturing Operation (FMO) to assure that matching components are fabricated from materials of the same heat where specified by the Engineering Development Subsection.

2. Perform the necessary fuel element process development work, as approved by the Engineering Development Subsection, to establish the fuel element fabrication process.

3. Prepare and obtain concurrence of the Engineering Development Subsection on the process specifications for the fabrication of the fuel elements. These process specifications will include the quality control program for fuel fabrication.

4. Fabricate the fuel elements for the initial EVESR loading in accordance with the process specifications as well as the designs and specifications provided by the Engineering Development Subsection.

5. Conduct the quality control program given in the process specifications and maintain the required quality control records.

6. Perform special measurements to be used later for comparison with similar measurements to be made at RML.

C. Expected Results: Delivery of satisfactory fuel elements on schedule and within budgeted costs.

Sub-task C-3: Neutron Source Fabrication

A. Objective: To provide the necessary neutron sources for the EVESR reactor start-up.

B. Approach to Problem: Prepare drawings, procure materials, fabricate, irradiate and inspect the four EVESR neutron sources. Deliver to EVESR site on schedule for start-up.

C. Expected Results: Activated neutron sources delivered to EVESR site in time for scheduled critical testing of the EVESR core.

Sub-task C-4: Fuel Evaluation Instrumentation

A. **Objective:** To provide the instrumentation required in addition to that supplied by the EVESR Project in order to perform the EVESR fuel performance evaluation tests for both the initial fuel and the advanced experimental fuel.

B. **Approach to Problem:**

1. Starting with the basic sensor and data requirements established by the Engineering Development Subsection and the existing project furnished instrumentation, proceed with detailed layout and design of a complete, compatible, instrument system that will provide the required test data. Cooperate with Engineering Development Subsection on design of instrument attachments to the fuel bundles.

2. Establish a design and procurement schedule which will assure reactor start-up and check-out on schedule (R & D instrument panel may be installed subsequent to reactor start-up). Provide written equipment design descriptions, maintenance and operating instructions for the EVESR Operations and Maintenance Manual.

3. Procure, assemble, install, test and calibrate the instrumentation.

C. **Expected Results:** Adequate and satisfactory instrumentation, in addition to that provided by the EVESR Project, to meet all requirements of the EVESR Superheat Fuel Development Program. All of the above work is to be completed within the budgeted funds shown in Section E below.

Sub-task C-5: Coolant Chemistry Sampling Equipment (Design & Installation) (Formerly Sub-task D-5)

A. Objective: To design, procure, and install the necessary sampling lines, equipment and instruments for use in connection with coolant chemistry and radiochemistry measurements and location of defected fuel bundles in the EVESR reactor core.

B. Approach to Problem:

1. Establish the types of coolant chemistry measurements necessary for the detailed evaluation of superheat fuel performance in EVESR. This includes determining what techniques and equipment are best suited for the collection of coolant chemistry data, for detection and location of defected fuel bundles, and for the measurement of coolant impurities and fuel activity release rates.

2. Review with Superheat Development Unit and obtain concurrence on the specifications and cost of the sampling equipment and instrumentation.

3. Submit the specifications to a design group for detailing.

4. Procure necessary equipment and arrange for and follow its installation.

5. Test and adjust the sampling equipment as necesssary to assure that equipment is in good working order for EVESR operation.

C. Expected Results: Installation of the necessary equipment and instruments to provide chemistry data for the evaluation of superheat fuel performance.

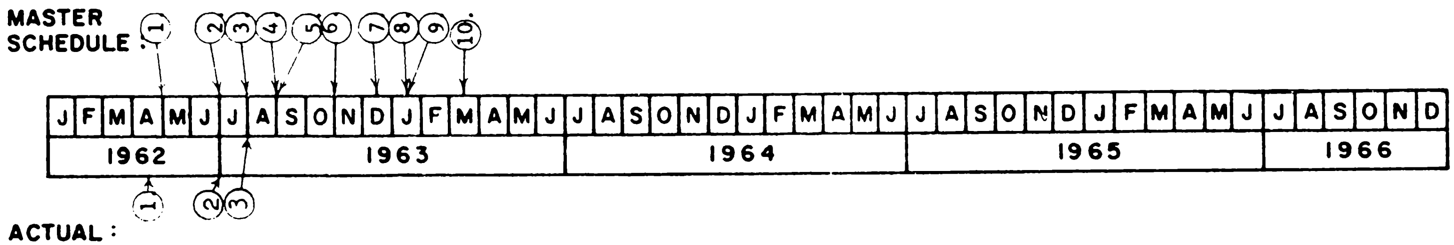
TASK C

FUEL PROCESS DEVELOPMENT,

INSTRUMENTATION & INSTALLATION

EVESR R & D PROGRAM

1. Start Process Development Activity
2. Start Instrumentation Design
3. Start Chem. Sample Sta. Design
4. Start Fabrication of Fuel Elements
5. Start Source Fabrication
6. Start Assembly Mark II Bundles
7. Source Irradiation Getr Cycle 40
8. Complete Fabrication of Fuel
9. Complete Source Irradiation
10. Load Source in EVESR



DATE : August 31, 1962

REV. 0 DATED: Aug. 31, 1962

Sub-task D-1: Fuel Tests, Engineering Physics

A. Objective: To provide the necessary engineering physics effort to carry out planning, evaluation, and operations assistance in connection with EVESR operation and physics tests with initial and advanced fuel through December 1965.

B. Approach to Problem:

1. Provide physics testing procedures for the fuel test program, including initial start-up and fuel reactivity measurements.

2. Evaluate physics data and provide data reduction for the fuel performance evaluations making available the nuclear performance parameters required to evaluate the thermal and hydraulic performance.

3. Provide assistance to the EVESR operations staff when required to assure that physics tests are performed in the prescribed manner. (Funds for such assistance are provided in Task D-2).

C. Expected Results: A well founded and performed physics test program yielding the necessary data to evaluate fuel performance.

Sub-task D-2: Fuel Tests - Engineering Assistance for Site

A. Objective: To provide the necessary on-site technical coordination and direction as required to carry out the detailed fuel test procedures and data collection.

B. Approach to Problem: Estimate requirements and provide assistance from other groups such as Engineering Development, Engineering Physics, Fuels and Materials Development, VAL Chemistry, VISO, etc., as required to:

1. Insure that basic data necessary for the fuel development program are obtained and properly recorded for direct key punching.

2. Coordinate and integrate fuel tests.

3. Provide on-site technical direction and interpretations of detailed test procedures.

4. Perform on-site evaluations of test results, and make recommendations for changes in test procedures when necessary to achieve desired results.

C. Expected Results: The expected results from this task are to provide experimental information on operating limits and failure modes for fuel elements irradiated in the fuel development program.

Sub-task D-3: Fuel Tests - Test Procedures

A. Objective: To provide detailed test procedures for use by EVESR operating group.

B. Approach to Problem:

1. Plan and specify in detail the tests and test conditions under which each phase of the test program will be conducted.

2. Write test operation manuals for the use of VISO personnel.

3. Collect test procedures from other performing components and coordinate effort of preparing test procedure manuals.

4. Provide graphs, nomographs or other types of operating aids.

C. Expected Results: Clearly written and easily understood directions concerning the manner in which each test phase is to be conducted, including test manuals issued well in advance of actual performance of tests as well as specific engineering procedure for all EVESR tests.

Sub-task D-4: Fuel Tests - Data Reduction

A. Objective: To provide a computer program or programs which can be used as a data reduction tool for converting raw process data into fuel performance parameters and as a tool for setting operating limits for the fuel test program.

B. Approach to Problem:

1. Develop the necessary analytical techniques for calculating fuel performance parameters such as fuel clad and central temperatures from raw test data.

2. Check analytical solution by conducting a hand calculated test case wherever it is practical.

3. Organize the solution for machine calculations.

4. Provide for, assist and follow the programming and debugging effort including the machine calculations of test cases to insure proper functioning of program.

5. Develop input and output routines for machine calculations. This includes developing appropriate raw data record sheets which can be used directly for packing cards for computer input, as well as developing the routine for converting the machine calculation output to an easily read and understood presentation format.

6. Follow initial data reduction runs to eliminate any remaining problem areas.

7. Provide for limited programming follow-up to improve running time of program.

C. Expected Results:

1. A computer program which can be used to:

   a. Reduce the large quantities of raw test data to useable fuel performance parameters, with a minimum of engineering effort.

   b. Set operating limits for the fuel test program.

2. Input and output routines which will process data from field generated records and print-out in a form which can be reproduced and included in progress reports directly.

Sub-task D-5: Fuel Activity Release and Coolant Chemistry
(Formerly Sub-task E-3)

A. Objective: To detect and locate failed fuel elements, to measure the radioactivity release rate from defected fuel elements, and to study the effect of coolant chemistry on the fuel cladding performance.

B. Approach to Problem:

1. Sample and make chemical, radiochemical and radioactivity measurements on the reactor coolant at appropriate points in the system to detect and locate failed fuel elements. Monitor all thirty-two bundle exits for presence of a failed fuel element.

2. Measure and evaluate the impurities present in the coolant reaching the fuel cladding. These evaluations will be repeated under various operating conditions.

3. Measure and determine radioactivity release rates from defected fuel.

4. Prepare and obtain Superheat Development Unit concurrence on annual chemistry development program covering the work proposed under this task.

5. Maintain records of coolant chemistry conditions for later use in evaluating fuel performance when post-irradiation examination data is available.

C. Expected Results:

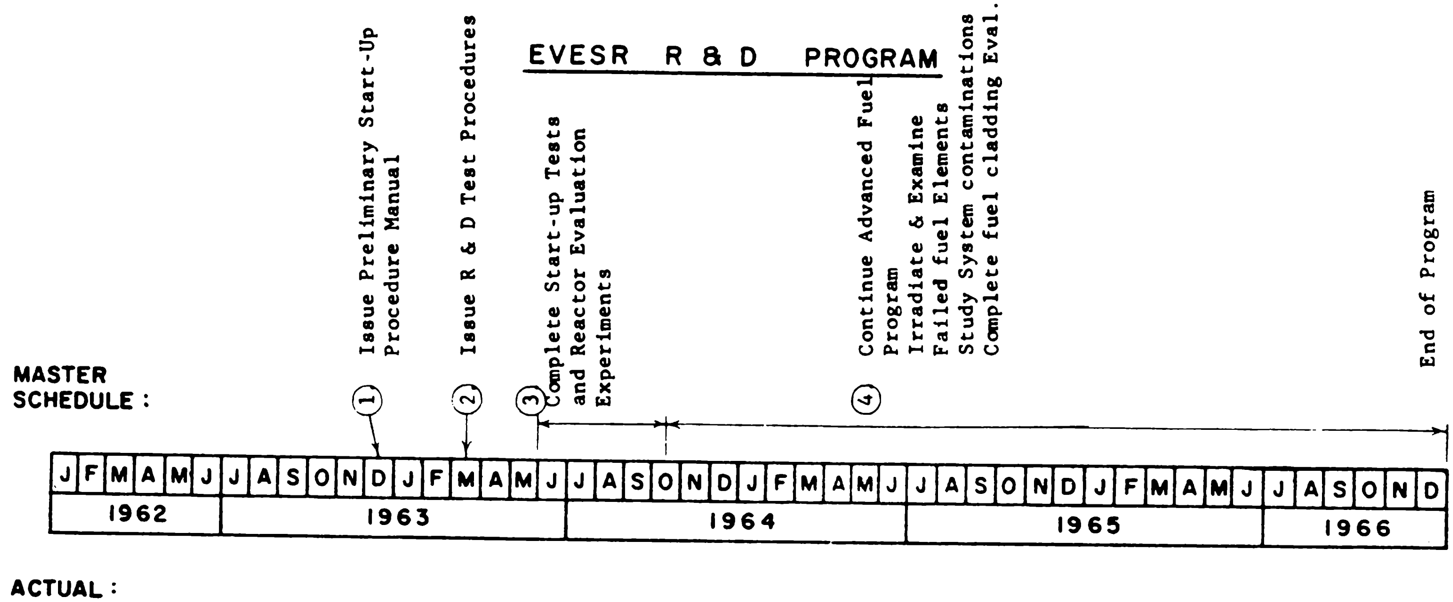
1. Determination of the coolant impurities present in the fuel coolant channels under various operating conditions.

2. Determination of the radioactivity release rate from fuel under various fuel performance levels.

3. Determination of the location of any fuel failures.

TASK D

FUEL TEST PROGRAM



DATE : August 31, 1962

REV. 0 DATED: Aug. 31, 1962

Sub-task E-1: Post-Irradiation Examination Planning and Evaluation

A. Objective: To plan the post-irradiation measurements to be made on EVESR initial and advanced fuel and to evaluate the results.

B. Approach to Problem:

1. Prepare and provide to RML, with the concurrence of the Superheat Development Unit, specifications, schedules and cost control for the post-irradiation examinations of EVESR fuel.

2. Follow the post-irradiation examination work at RML to assure that it is performed on schedule, as prescribed and within budgeted costs.

3. Evaluate the data from all post-irradiation examinations and provide letter reports to the Superheat Development Unit accompanied with RML raw data.

C. Expected Results: Well conceived and performed post-irradiation examinations yielding the best obtainable data from irradiated EVESR fuel.

Sub-task E-2: Pre- and Post-Irradiation Examinations

A. Objective: To determine the irradiation effects on selected superheat fuel elements irradiated in EVESR to establish power distribution, physical and dimensional changes, metallurgical changes, and cause of any failures.

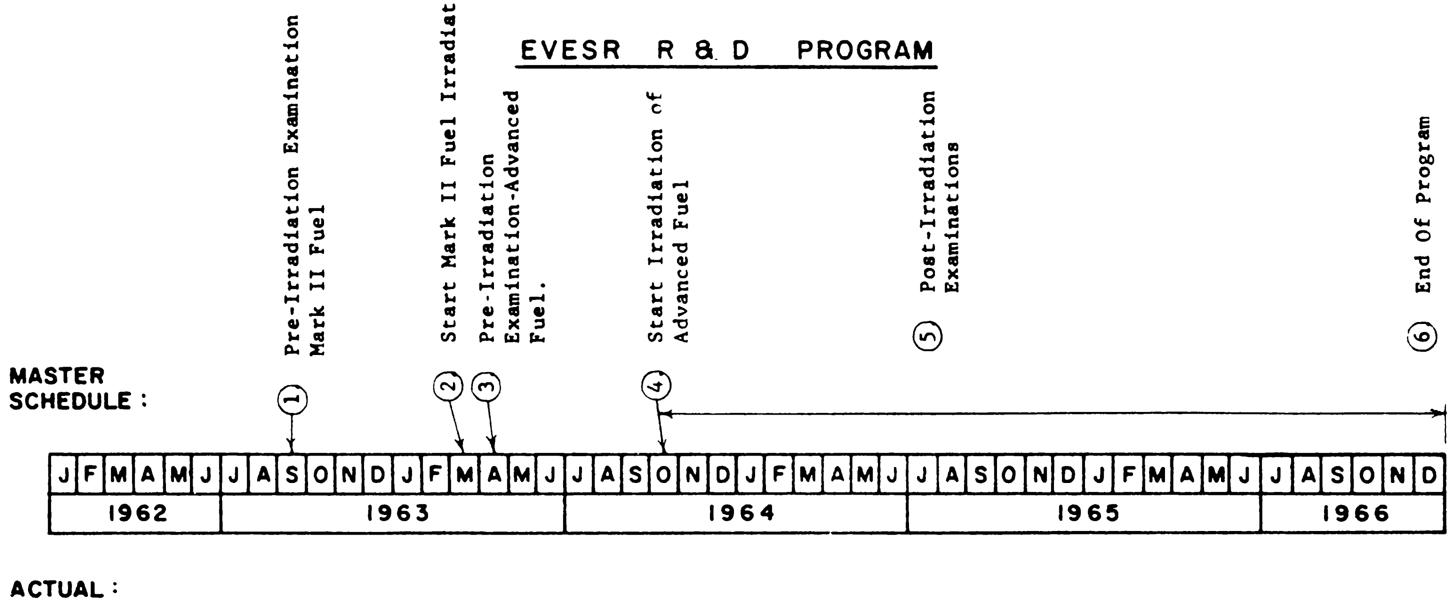
B. Approach to Problem: Perform pre- and post-irradiation examinations on sound, purposely defected, and in-service failed fuel elements in order to establish failure mode and propagation of failure mechanism of superheat fuel elements under actual service conditions. All examinations are to be performed in accordance with specifications prepared by Fuels and Materials Development and approved by the Nuclear Superheat Development Unit.

C. Expected Results: Accurate data which will permit determination of the significance of design variables on superheat fuel performance, determination of the significance of fuel burnup on superheat fuel performance, and insight into mode of failure which will permit improved fuel design.

TASK E

FUEL POST-IRRADIATION
EXAMINATIONS

EVESR R & D PROGRAM



DATE : August 31, 1962

REV. 0 **DATED:** Aug. 31, 1962

Sub-task F-1: Advanced fuel Design and Process Development

A. Objective: To provide the complete and detailed mechanical, thermal, hydraulic and physics analyses, designs and design specifications for the fabrication of the EVESR advanced experimental fuel.

B. Approach to Problem:

1. Establish all design requirements and design objectives for the advanced fuel, including the mechanical configuration, the materials of fabrication, and performance parameters and limits.

2. Evaluate the superheat fuel performance data resulting from ESADE and EVESR irradiation experience, as well as any other applicable experimental data that becomes available, and utilize these data in the advanced fuel designs.

3. Conduct the necessary design studies and analytical investigations and complete the detailed designs of advanced superheat fuel elements which have potential for improving neutron economy, superheat reactor thermal efficiency and core fabrication costs Make suitable computer codes for evaluating the advanced fuel based on the experimental data that will be obtained during its irradiation in EVESR.

4. Obtain assistance from other technical and laboratory groups in areas such as physics, materials, laboratory testing, fabrication process development, etc. as required to complete the advanced fuel designs and produce shop drawings and specifications for the fabrication of the advanced fuel.

C. Expected Results: Completion of designs and specifications which will permit fabrication of advanced fuel of economic interest - on schedule and within the budgeted funds.

Sub-task F-2: Advanced Fuel Fabrication

A. Objective: To provide overall control and responsibility for the fabrication of the EVESR advanced experimental fuel.

B. Approach to Problem:

1. Assign responsibilities to performing components for the various phases of purchasing inspection, fabrication, assembly and quality control for the advanced fuel.

2. Provide all necessary specifications, drawings and engineering instructions for the above work.

3. Maintain cost and schedule control in all areas.

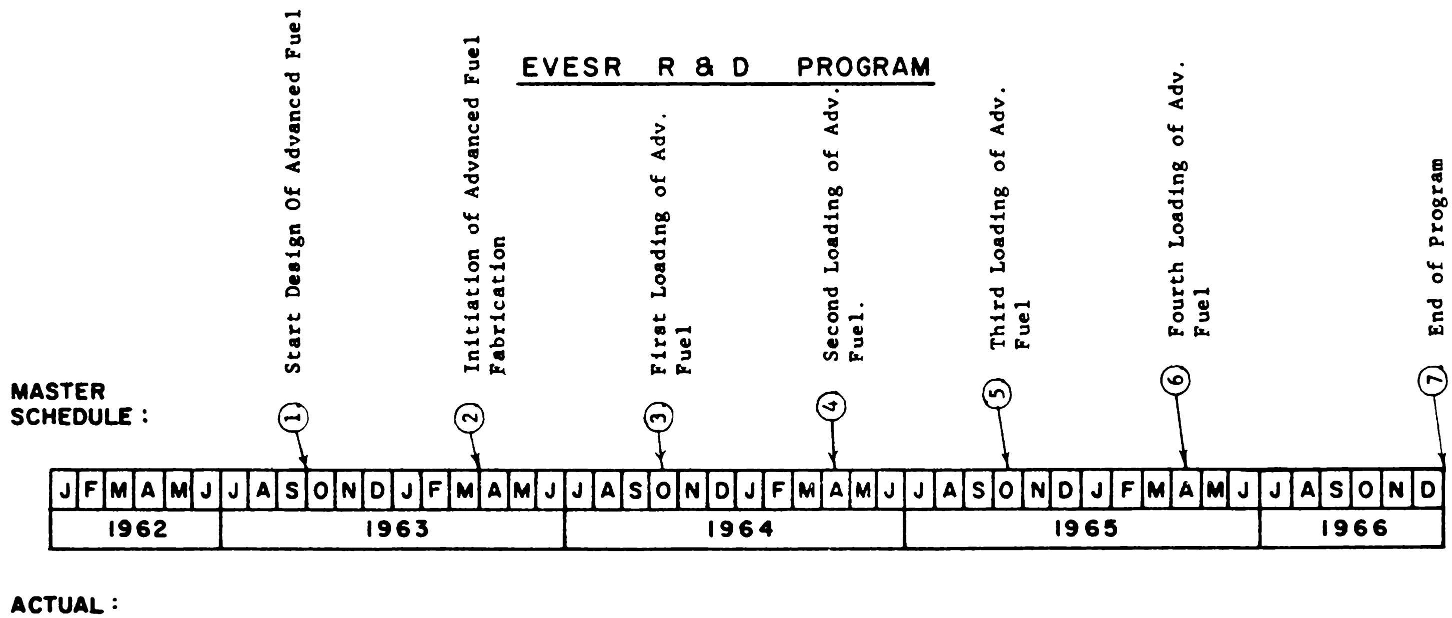
4. Follow progress of fuel fabrication through shops to assure compliance with all drawings and specifications (fabrication following).

5. Measure cost and schedule estimates against actual performance.

C. Expected Results: Delivery of advanced EVESR fuel elements on schedule and within budgeted costs.

TASK F

ADVANCED FUEL PROGRAM



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REV. 0 DATED: Aug. 31, 1962