

1. Considered April 23, 1958, Appl. No. 519-3
2. Name of Applicant Northern States Power Company
3. Post Office Address Sioux Falls, South Dakota
4. Amount of water claimed 6.0 cfs
5. Source of water supply Big Sioux River
6. Water to be used for Industrial use
Acres to be irrigated _____
7. Water to be diverted and used in Minnehaha County about
6-7 miles northeast of Sioux Falls.
8. Proof of Publication Received ☒
Not Received _____
9. Application Approved _____
Subject to _____
Not approved _____
Action deferred _____
10. Remarks:

No. 519-3
 Water Division No. 3 Big Sioux River District 1
 (Blanks to be filled by the Engineer for the W. R. C.)

APPLICATION FOR PERMIT To Appropriate Water within the State of South Dakota

(NOTE—Draw a line through items not applicable.)

1. Name of applicant W. N. Marx, Manager, Sioux Falls Division
 Postoffice address Sioux Falls, County Minnehaha, State So. Dak.

I. If a corporation

(a) Name of same Northern States Power Company of Minnesota
 (b) Date and place of Incorporation _____
 (c) Amount of capital stock Adequate
 (d) Amount paid in Adequate
 (e) Names and addresses of directors See attached list.

(NOTE—A certified copy of articles of Incorporation must accompany the application.)

II. Method of accomplishing the work and financial resources of the applicant:

(a) Method of accomplishing work (Whether by contract, employment of others, or by direct labor)
contract

(b) Cash on hand, \$ adequate (c) Treasury stock, \$ _____
 (d) Bonds to be issued, \$ _____ (e) Other resources, \$ _____

2. Name of diversion work Pump House for Power Plant
 3. Amount of water claimed 6.0 cubic feet per second
 4. Source of water supply Big Sioux River
 5. Location of point of diversion See attached map—Approx. center of SE 1/4 of
Sec. 30, T102N, R48W on right bank Minnehaha County.
 6. Annual period or periods during which water is to be used Throughout the year
 7. To be used for:

I. Irrigation or domestic use: Gravity, overhead sprinkling or combination system?

(a) Number of acres to be irrigated _____ acres.

(b) Legal subdivisions to be irrigated See attached map II (a) b. l. c.

(NOTE—A list of lands to be irrigated, giving each subdivision and fraction with acreage thereof, should be written here, or may be appended as a part of this application. Same must also be shown on accompanying map.)

(c) Statement as to domestic use (giving location, etc.) _____

II. Stockwatering, mining, milling, power, fish culture, fire protection and public recreation:

(a) Nature of use Electric power production - cooling

Name any of the above uses claimed

(b) Amount of power to be generated 88,500 horse power.

Amount claimed for engine cooling 6.0 cfs

(c) Location of plant Adjacent to south line of SE 1/4 Sec. 30, T102N, R48W
Minnehaha County

(d) Method of developing power Steam turbine-generator

(e) Point where water will be returned to stream See attached map—

SE 1/4 SE 1/4, Sec. 30, T102N, R48W, Minnehaha County

Note: Approx. 3.5 cfs returned to river—balance lost by
evaporation.

Map No. 398-3

8. Estimated cost of work:

(a) Head Gates, \$

(c) Fluming, \$

(e) Other structures

Pumping plant, \$ 2,000
Total, \$ 300,000

9. Description of diversion works.

I. Nature of works: (Reservoir, dam, ditch, flume, canal, etc.)

Pumping plant - detailed plan is not yet completed

II. Dimensions of works:

(a) Dam: Height feet; length at bottom

feet; thickness at bottom

feet; slope at front water face

material used in construction

(b) Reservoir: Capacity when filled

acres. Surface area at water mark

DEPTH AT OUTLET-Feet	SURFACE AREA AT EACH DEPTH AT OUTLET-ACRES	FEET	ACRE-Feet CAPACITY

(c)* Head gate: Width

Material

(d) Canal: Total length

LOCATION BELOW HEADGATE	DEPTH	BOTTOM WIDTH	WIDTH AT WATER-LINE	GRADE PER MILE
At Mile	feet	feet	feet	feet
At Mile	feet	feet	feet	feet
At Mile	feet	feet	feet	feet
At Mile	feet	feet	feet	feet
At Mile	feet	feet	feet	feet
At Mile	feet	feet	feet	feet

(Give dimensions where reductions in size are made.)

10. Time required for completion of work five years.

11. Time required for complete application of water to the proposed beneficial use four years.

12. Choice of newspaper for publication of notice of intention to appropriate Sioux Falls
Argus-Leader

STATE OF SOUTH DAKOTA

Minnehaha

County of

W. N. Marx

I, W. N. Marx, being first duly sworn on my oath depose and say: That my relation to the above described undertaking is that of owner, that I have read the above foregoing statement, examined the map accompanying the same, and that I know of my own personal knowledge that the matters therein stated and shown are true.

Signed

Manager, Sioux Falls Division

Subscribed and sworn to before me this

6th

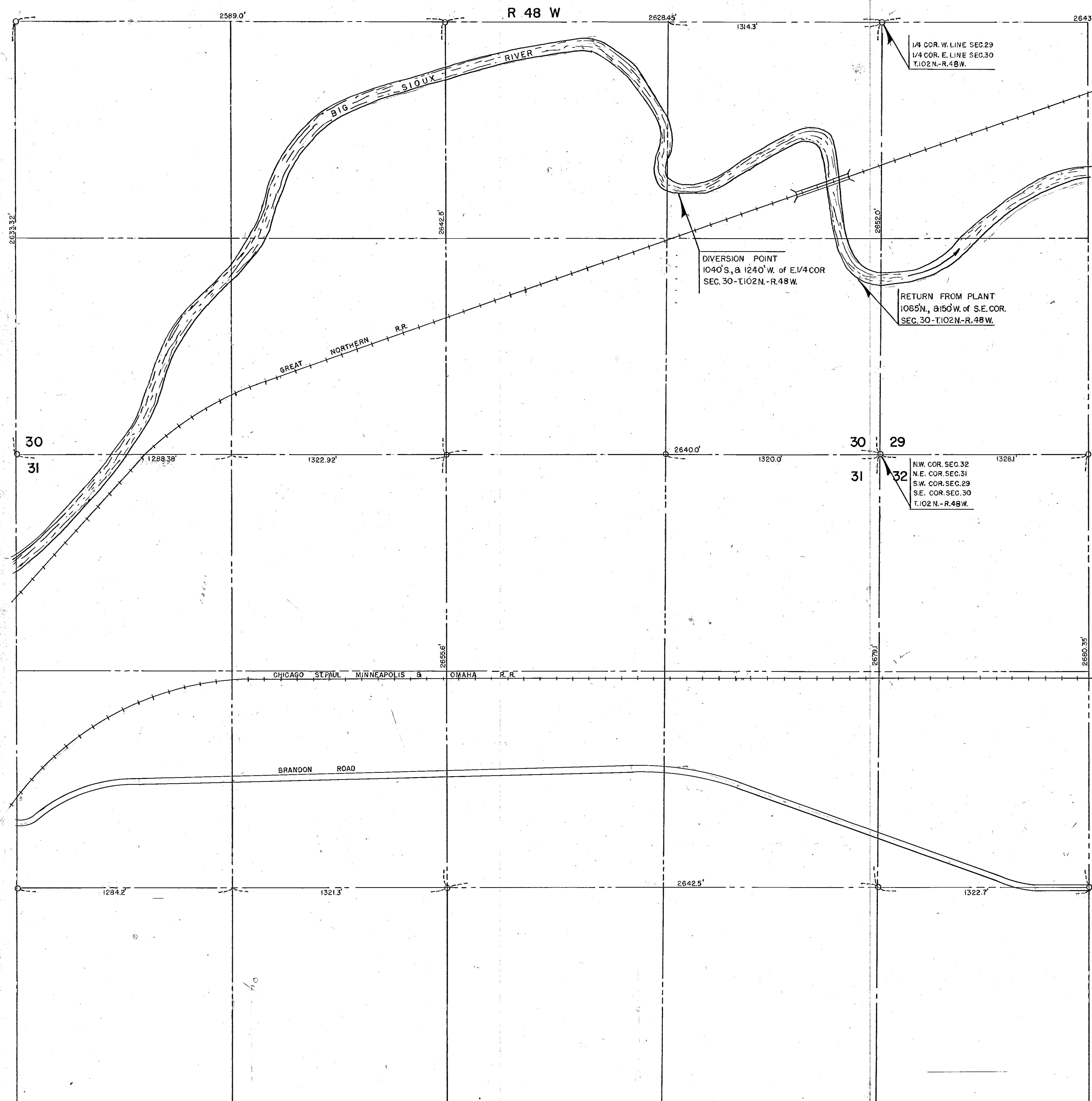
day of

February

1958

Notary Public (or other qualified officer.)

*Each diversion must have a headgate or measuring device.



NOTE: O Indicates Marked Cor.

SURVEYORS CERTIFICATE

I, K.E. BENSON, Registered Land Surveyor and Professional Engineer of the State of SOUTH DAKOTA Do
Hereby Certify That I have Mapped the Area as Shown For Purposes of Water Appropriation For Industrial
Use.

K.E. Benson
K.E. BENSON

STATE OF SOUTH DAKOTA
COUNTY OF MINNEHAHA

On this 8 Day of Feb. 1958, Before Me, the Undersigned, A Notary Public, Appeared
K.E. BENSON, well known to me to be the Person Described above and who Acknowledged to me that he Executed
Foregoing Certificate.

John W. 5-1960
My Commission Expires

John W.
Notary Public

N.S.P. CO.
PATHFINDER STEAM PLANT
DIVERSION No. 1
MINNEHAHA COUNTY, S. DAK.

Map No. 398-3
(Appl. No. 519-3)

Remarks by State Water Resources Commission:

STATE OF SOUTH DAKOTA
County of Hughes } ss.

Pierre, South Dakota, February 14, 1958

This is to certify that the foregoing application was received by the Commission at 8:15 o'clock
9 a.m. upon the 10 day of February, 1958

State Water Resources Commission

By:

Chief Engineer, Executive Officer

Number of permit 579-J

Date of first receipt of application February 10, 1958

Date of return to applicant for correction, amendments or changes required as follows: February 10, 1958

Date of receipt of corrected application February 14, 1958

Date from which applicant may claim right February 10, 1958

Approved April 23, 1958, Recorded in Book 4 Page 228

This is to certify that we have examined the foregoing application for a permit to appropriate water of the State of South Dakota, and we hereby grant the same as stated herein, subject, however, to the following limitations and conditions:

1st. The equivalent of at least one-fifth of the work above specified is to be completed on or before

October 23, 1960

2nd. The whole of said work is to be completed on or before April 23, 1963

3rd. The limit of time for proof of beneficial use of water appropriated in accordance herewith is

April 23, 1967

4th. The water appropriated shall be used for the purpose industrial use

5th. The prior right of all persons who, by compliance with the laws of the State of South Dakota, have acquired a right to the use of water must not be injuriously affected by this appropriation.

6th. The amount of the appropriation herein granted shall not exceed 6.0 cubic feet per second; neither shall it exceed the capacity of the above described system of diversion works, nor the least amount of water that experience may hereafter indicate as necessary for the production of crops in the exercise of the best husbandry, and further, said appropriation must be limited to not more than one-seventieth (1/70) of one cubic foot of water per second of time for each acre of land to which water is actually industrial use, used and beneficially applied for irrigation on or before April 23, 1967; said water to be used during the following described annual period:

Continuously

Witness my hand this 5 day of May, 1958

State Water Resources Commission

By:

Chief Engineer, Executive Officer

Certificate of Construction Issued August 16, 1963

Water License Issued August 16, 1963

DIRECTORS - NO. NERN STATES POWER COMPANY

NAME

ADDRESS

H. C. Cummins	Minneapolis, Minnesota
Allen S. King	Minneapolis, Minnesota
F. W. Anderson	Eau Claire, Wisconsin
H. M. Gallagher	Waseca, Minnesota
R. F. Pack	Minneapolis, Minnesota
W. S. Moscrip	St. Paul, Minnesota
T. G. Harriscn	Minneapolis, Minnesota
A. B. Jackson	St. Paul, Minnesota
B. C. Marks	Fargo, North Dakota
P. H. McDowell	Sioux Falls, South Dakota
S. P. Duffy	Minneapolis, Minnesota

STATE OF SOUTH DAKOTA

WATER LICENSE No. 519-3

WHEREAS, On the 10th day of February A. D. 1958
the Northern States Power Company

of Sioux Falls County of Minnehaha and State of
South Dakota duly made application (No. 519-3) to us for a permit to use 6.0
cubic feet per second of the waters of the Big Sioux River
County of Minnehaha, State of South Dakota, for industrial
purposes; and

WHEREAS, On the 5th day of May A. D. 1958
Permit No. 519-3 was issued to said applicant for the diversion of said water, and providing for the
completion of the works of diversion therein described on or before the 23rd day of
April A. D. 1963 and for the application to beneficial use of said
water on or before the 23rd day of April A. D. 1967
and,

WHEREAS, On the 15th day of July A. D. 1963
the holder of said permit duly made proof of the completion of adequate works for the diversion of 6.0
cubic feet per second of said waters, as evidenced by our Certificate No. 519-3
dated August 16, 1963 confirming the completion of works of sufficient capacity for
diverting and conveying to the place of intended use 6.0 cubic feet per second of water, with date of
priority of February 10, 1963 and,

WHEREAS, On the 15th day of July A. D. 1963
proof was duly made of the application to beneficial use of 6.0 cubic feet per second of said water;
for cooling generators in Pathfinder power plant

NOW, THEREFORE, By the virtue of the authority vested in us by the laws of the State of South Dakota,
We hereby grant and confirm to the Northern States Power Company
of Sioux Falls the holder and owner of said permit No. 519-3
a water right, dating from February 10, 1958 to the use of 6.0
cubic feet per second of the waters of the Big Sioux River
in the County of Minnehaha and State of South Dakota, or so much thereof as may be
necessary for the purposes hereinafter mentioned, to be diverted at a point 1040'S and 1240'W of the
SE corner of Section 30, T102N, R43W
and conduct ~~the same~~ the power plant

for the purpose of cooling power generators

subject, however, to the laws of the State of South Dakota applicable to a license for the use of the waters of the
State, and subject, also to the local or community customs, rules and regulations which have been or may be
adopted from time to time by a majority of the users from a common source of supply, canal or lateral from
which such water may be taken, when such rules and regulations have for their object the economical use of such
water.

WITNESS, My hand and seal of our office at Pierre, South Dakota,
this 16th day of August A. D. Nineteen
Hundred and Sixty-Three

State Water Resources Commission

By:

J. W. GRIMES
Chief Engineer, Executive Officer

STATE OF SOUTH DAKOTA

Water License No. -----

To -----

Source of Supply -----

County -----

Am't ----- c. f. s.

Point of Diversion -----

Place of Use -----

Purpose -----

Date of Priority -----

Recorded -----

In Book ----- of ----- Page -----

Permit No. 519-3Water Division No. 3 Big Sioux River
Water District

Certificate of Completion of Works

TO ALL WHOM IT MAY CONCERN:

This is to Certify, That the Northern States Power Companyof Sioux Falls, County of Minnehaha, and State ofSouth Dakota, the holder of Permit No. 519-3, issued upon ApplicationNo. 519-3, bearing date of priority of February 10, 1958, authorizing the diversion of6.0 cu. ft. per second of the waters of the Big Sioux RiverCounty of Minnehaha, State of South Dakota, at a point 1040'S and 1240'W
of the E $\frac{1}{2}$ corner, Section 30, T102N, R46Wfor industrial

purposes, has fully complied with the provisions of the laws of the State of South Dakota relating to proof of completion of the works of diversion set out and described in said Permit; that said works are found in satisfactory condition for diverting and conveying to the place of intended use 6.0 cu. ft. per second of water; and that the ~~lands proposed to be irrigated by the~~ use of said water are described as follows, to-wit: to be used for cooling of power generators.

Witness my hand this 16th day of August A. D. 1963

State Water Resources Commission

By: J. W. GRIMES
Chief Engineer, Executive Officer

(First Publication before March 21, 1958.)**APPROPRIATION OF WATER**

Office of State Water Resources Commission.

Pierre, S. Dak., February 14, 1958.

Notice is hereby given that Sioux Falls Division, Northern States whose postoffice address is Sioux Falls S. Dak., has made an application in accordance with the provisions of the water laws of South Dakota for a permit to appropriate for beneficial use 6.0 cubic feet of water per second of time from the Big Sioux River through the Power Plant, the point of diversion of which is to be located on the southerly bank of said stream in the NE $\frac{1}{4}$ of the SE $\frac{1}{4}$ of Section 30 Twp. 102N Range 48W. Said water to be used for the purpose of providing irrigation on the following described land:

Industrial use in the SE $\frac{1}{4}$ Sec 30, T 102N, R 48W

XXXXXX
 This application will be taken up by the State Water Resources Commission at their office at Pierre for consideration upon the 23 day of April, 1958, at 2:00 p.m. All interested persons may appear and be heard.

Appropriate action will be taken by the Water Resources Commission after suitable time has elapsed for the consideration of any or all information presented.

State Water Resources Commission

By:

~~State Water Resources Commission~~
Les Wyborny
 Water Rights Engineer

PROOF OF PUBLICATION

STATE OF SOUTH DAKOTA }
County Minnehaha } ss.

**NOTICE OF INTENTION TO AP-
PROPRIATION OF WATER**
Office of State Water Resources Com-
mission, Pierre, S. Dak. February 14, 1958.
Notice is hereby given that Sioux Falls
Division, Northern States Power Company,
of Minnesota, whose postoffice address is
Sioux Falls, S. Dak., has made an ap-
plication in accordance with the pro-
visions of the water laws of South Da-
kota for a permit to appropriate for bene-
ficial use 6.9 cubic feet of water per sec-
ond of time from the Big Sioux River
through the Power Plant, the point of
diversion of which is to be located on
the southerly bank of said stream in the
NE 1/4 of the SE 1/4 of Section 30 Twp.
102N Range 48W. Said water to be used
for the purpose of industrial use in the
SE 1/4 Sec. 30 T 102N R 48W.
This application will be taken up by
the State Water Resources Commission
at their office at Pierre for consideration
upon the 23rd day of April, 1958, at 2:00
p.m. All interested persons may appear
and be heard.
Appropriate action will be taken by the
Water Resources Commission after suit-
able time has elapsed for the considera-
tion of any or all information presented.
—State Water Resources
Commission.
By: Les Wyburny,
Water Rights Engineer.
(Feb. 19, 23, March 4, 11)

A. A. ROESLER being duly
sworn, deposes and says the annexed printed copy of Notice
of Intention to Appropriate Water was taken from the _____
SIoux FALLS ARGUS LEADER a newspaper which
during the whole time of publication of said notice herein
after stated, has been and is printed and published in the
_____ City _____ of Sioux Falls _____ County of
Minnehaha _____ and State of South Dakota;
that the said notice was published in said newspaper on the
following dates:

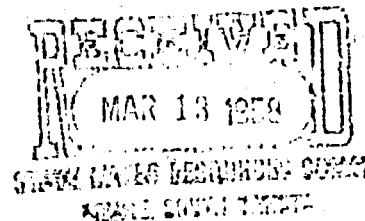
February 18, 1958

February 25, 1958

March 4, 1958

March 11, 1958

in each and every issue of the full number thereof, the first
publication being made on the 18th day of February
19 58 and the last publication on the 11th day of
March 19 58, upon which days or times
of publication aforesaid the newspaper was regularly pub-
lished, and that during the whole time of said publication he
was Comptroller the printer and publisher of
the said newspaper.



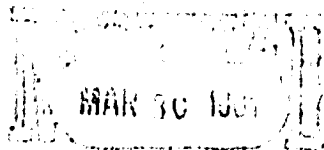
Subscribed and sworn to before me this 11th day of March A. D. 19 58

Notary Public _____ Minnehaha County, S. D.

Permit No. _____

Water Division No. _____

Water District _____



STATE WATER RESOURCES COMMISSION

NOTICE OF COMPLETION OF WORKS

This blank must be filled out by the holder of permit and forwarded to the Executive Officer at least 60 days before the time set for completion of works

(P. O.) Sioux Falls, South Dakota(Date) March 30, 1961

TO THE STATE WATER RESOURCES COMMISSION
OF SOUTH DAKOTA, PIERRE:

DEAR SIR: Notice is hereby given of the completion of the works for the diversion of 6.0
_____ cu. ft. per second of the waters of the state from Big Sioux
River
through the pump house
for electric power production - cooling purposes

_____ In accordance with the terms and conditions of a certain permit heretofore issued by the Water Resources Commission of the State of South Dakota.

1. The name of the person or corporation holding said permit is Northern States
Power Company

2. The postoffice address of such person or the place of business of such corporation is Sioux Falls
County of Minnehaha State of South Dakota

3. The number of such permit is 519-3 and the date set for the completion of such work is
June 1, 1961

4. Said water to be used for electric power production - cooling purposes.

5. Said works of diversion will be fully completed on the date set for such completion, and the amount of water which said works are capable of conducting to the place of intended use, in accordance with the plans accompanying the application for such permit, is 6.0 cubic feet per second.

6. The amount of land for which said water is available is _____ acres,
particularly described as follows: _____

I hereby certify that the facts set forth in the above notice are true.

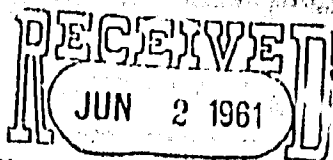
Northern States Power Company

(Signature of Owners)

Mark [Signature]
Manager, Sioux Falls Division

Permit No.

Water Division No.

Water District **STATE WATER RESOURCES COMM.**
PIERRE SOUTH DAKOTA

Notice of Application of Water to Beneficial Use

(This blank must be filled out by holder of permit and forwarded to the State Water Resources Commission at least sixty days before the time set for application of water.)

(P. O.) Sioux Falls, South Dakota(Date) May 31, 1961

TO THE STATE WATER RESOURCES COMMISSION OF SOUTH DAKOTA, PIERRE:

DEAR SIR: Notice is hereby given that on the first day of August
1961, at Sioux Falls County of Minnehaha State of South Dakota,
before _____ (Examining Engineer or Officer) proof will be submitted of the application to beneficial use
of 6.0 cubic feet per second of the waters of Big Sioux River

_____ In accordance with the terms and conditions of Permit No. 519-3
heretofore issued by the Water Resources Commission of South Dakota.

1. The name and postoffice address of the person or corporation holding said permit are Northern States Power Company
2. The use to which said water has been applied is electric power production - cooling.
3. The amount applied to beneficial use is 6.0 cubic feet per second
4. The place where said water is used is (if for irrigation, give full and accurate description of the lands irrigated) Pathfinder Atomic Power Plant
5. The name of the canal or ditch or other works by which said water is conducted to such place of use is pump house.
6. The right to take the water from such works is based upon permit No. 519-3
7. The source of supply from which such water is diverted is Big Sioux River.
8. The date of priority which said user is prepared to establish is February 10, 1958

Northern States Power Company
(Sign) [Signature]
Manager, Sioux Falls Division

Permit No. 519-3Water Division No. 3Water District 1

Certificate of Examination of Works

Report of Examining Engineer

To the State Water Resources Commission, Pierre, South Dakota:

Dear Sirs: I hereby certify that in accordance with your letter of authorization dated 10-10-58, I have this day made a thorough examination of the diversion works constructed by NORTH DAKOTA STATE POWER COMPANY, of SILOSVILLE, MINNEAPOLIS County, South Dakota, holder of Permit No. 519-3, issued upon Application No. 519-3, bearing date of priority of FEBRUARY 10, 1958, authorizing the diversion of 1.2 cu. ft. per second of the waters of Big Sioux River for industrial purposes. I have to report on the condition of the same as follows:

The diversion works comprise: 2 ALLIS CHALMERS 12" PUMPS #15VTK, POWERED BY ALLIS CHALMERS 100HP ELEC. MOTOR #H5MGI, OPERATING @ 1700 RPM OR 2000 GPM SETTING, 2-10" PIPES FROM PUMPS TO 12" MAIN TO MASTER CONTROL VALVE WHERE IT DIVIDES INTO 2-10" LINES TO 2 PRECIPITATORS THEN TO "CLEAR WELL" TO COOLING TOWER and the system is in the following condition: NEW

The point of diversion is located 1040' S E 1240' W of the E 1/4 CORNER, SECTION 30, T102N, R48W

The works are capable of diverting and conveying to the place of use 8.88 cu. ft. per second of water, which is to be used for COOLING OF ELECTRICAL GENERATORS

The legal sub-divisions to be irrigated are: _____

_____ comprising a total of _____ acres of land.

The domestic use intended comprises _____

The utilization for power, mining, or other beneficial uses comprises: _____

Witness my hand, this 15th day of July

A. D. 19 63

—SEAL—

Examining Engineer

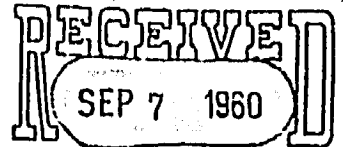
officer

NSP

NORTHERN STATES POWER COMPANY

SIoux FALLS, SOUTH DAKOTA

September 1, 1960



STATE WATER RESOURCES COMM.
PIERRE, SOUTH DAKOTA

Mr. J. W. Grimes,
Executive Office and Chief Engineer
Water Resources Commission
Pierre, South Dakota

Dear Mr. Grimes:

We are enclosing your copy (#81) of Amendment
No. 6 covering the Supplement No. 4 to the Safeguards
Report (ACNP-5905) for our application for the Pathfinder
Atomic Power Plant.

Yours very truly,

[Signature]
Manager

W.H.Rork
DEH/dh

Encl.

BEFORE THE
UNITED STATES
ATOMIC ENERGY COMMISSION

AMENDMENT NO. 6

TO
APPLICATION FOR AEC TYPE 104b LICENSE AND CONSTRUCTION
PERMIT FOR A UTILIZATION FACILITY INVOLVED IN THE
CONDUCT OF RESEARCH AND DEVELOPMENT ACTIVITIES LEADING
TO THE DEMONSTRATION OF THE PRACTICAL VALUE OF THE
FACILITY FOR COMMERCIAL PURPOSES

Northern States Power Company, a corporation organized under the laws of the State of Minnesota, applicant for an operating license and construction permit for the Pathfinder Atomic Power Plant, hereby amends Part II of its application to include Supplement No. 4 to the Safeguards Report (ACNP-5905). Supplement No. 4 consists of insertion pages and illustrations, dated August 15, 1960, and instructions explaining the insertion of these pages and illustrations into the Safeguards Report (ACNP-5905).

This amendment to the application for a license contains no restricted or other defense information.

IN WITNESS WHEREOF the applicant has caused its name to be hereunto signed by Earl Ewald, its Vice President in Charge of Operation, and its corporate seal to be hereunto affixed by M. A. Morrissey, an Assistant Secretary, thereunto duly authorized, this 24th day of August, 1960.

NORTHERN STATES POWER COMPANY

By /s/ Earl Ewald
Earl Ewald
Vice President
in Charge of Operation

ATTEST:

/s/ M. A. Morrissey
M. A. Morrissey
Assistant Secretary

(Corporate Seal)

STATE OF MINNESOTA }
COUNTY OF HENNEPIN } ss.

EARL EWALD, being first duly sworn, says that he is the Vice President in Charge of Operation of Northern States Power Company, a corporation organized and existing under and by virtue of the laws of the State of Minnesota, that he executed the foregoing amendment for the purpose therein set forth, that the statements made in said amendment are true and correct to the best of his knowledge and belief, that he was authorized to execute said amendment on behalf of said corporation, and that the seal affixed is the corporate seal of said corporation.

/s/ Earl Ewald

Earl Ewald

Subscribed and sworn to before
me this 24th day of August, 1960.

/s/ Pearl S. Duggan

PEARL S. DUGGAN
Notary Public, Hennepin County, Minn.
My Commission Expires June 23, 1964.

(Notarial Seal)

SUPPLEMENT NO. 4
to
Preliminary Safeguards Report
Pathfinder Atomic Power Plant

Prepared under AEC Contract No. AT(11-1) - 589
dated 22nd Day of November 1957
between
Northern States Power Company
and
U. S. Atomic Energy Commission

by

Northern States Power Company
15 South Fifth Street
Minneapolis 2, Minnesota

Approved: Hibbert Hill

Hibbert Hill
Chief Engineer

Approved: Earl Ewald

Earl Ewald
Vice President
in Charge of Operation

August 15, 1960

August 15, 1960

SUPPLEMENT NO. 4
to
SAFEGUARDS REPORT NO. ACNP-5905
on
PATHFINDER ATOMIC POWER PLANT

INSTRUCTIONS FOR REMOVAL OF SUPERSEDED SHEETS AND INSERTION OF NEW SHEETS	
Remove Sheets Listed Below Previously Dated	Insert Sheets Listed Below Dated August 15, 1960
I & II	I
III	II & III
IV, V, VI, VII, VIII	IV, V, VI, VII, VIII, IX
IX, X, XI, XII	X, XI, XII, XIII, XIV
1.1. thru 1.30	1.1 thru 1.85
Figs. 1.1 thru 1.11	Figs. 1.1 thru 1.25
3.1 thru 3.14	3.1 thru 3.14
Figs. 3.1, 3.2	Figs. 3.1, 3.2
Figs. 3.4, 3.5, 3.6, 3.7	Figs. 3.4, 3.5, 3.6, 3.7, 3.8
6.13 & 6.14	6.13 thru 6.14
7.10 & 7.11	7.10 thru 7.11
B1 thru B7	(Superseded by Section I)
Fig. B1	

Note: Text for Sections I and 3 are printed on both sides of each sheet, since these sections are complete.

August 19, 1960

I INTRODUCTION

This Safeguards Report describes the Pathfinder Atomic Power Plant; its site, equipment, and operation. The report is based on correlated contributions from the Northern States Power Company, Allis-Chalmers Manufacturing Company and their consultants. It is intended that this report should present the plant design philosophy and site description in detail. As design progresses, the report will be amended so as to result in a Final Safeguards Report.

Owners and operators of the Pathfinder Atomic Power Plant will be the Northern States Power Company of Minneapolis, Minnesota. Central Utilities Atomic Power Associates (CUAPA), a group of ten private midwest utilities, and the U. S. Atomic Energy Commission, under the provisions of Contract No. AT-(11-1)-589, contribute toward the associated research and development program. The plant, including the reactor, will be designed and constructed by the Allis-Chalmers Manufacturing Company of Milwaukee, Wisconsin. The Pioneer Service and Engineering Company of Chicago, Illinois will provide the architect and engineering services for the project.

The Pathfinder Atomic Power Plant will be located on a site on the south bank of the Big Sioux River, 5.5 miles northeast from the center of Sioux Falls, South Dakota. The conceptual design of the plant is shown in Figures I, II and III. It is scheduled for operation in June, 1962.

A research and development program, as provided for in Appendix B of the AEC Contract No. AT-(11-1)-589, is in progress at the facilities of the Allis-Chalmers Manufacturing Company and its subcontractors. A critical experiment facility (see ACNP-5809) provides means for determining and checking nuclear constants in a series of experiments. Corrosion and heat transfer test loops provide a verification of engineering data.

August 15, 1960

II SUMMARY

Presented in this report is a description of the design and operation of the Pathfinder Atomic Power Plant and its reactor with nuclear superheater. The plant site and the safeguards are reviewed in detail.

This report substantiates that the health and safety of the public and operating personnel will be protected by proper plant design and safe operating procedures. Various types of accidents are analyzed in order to verify that adequate safeguards are provided.

The reactor for the plant is of the heterogeneous boiling water type with a two-region core. A superheater region in the center is approximately a right circular cylinder surrounded by a boiler region. The fuel elements in the superheater region will contain highly enriched uranium dioxide in a stainless steel cladding with stainless steel. The fuel elements in the boiler region will contain slightly enriched uranium dioxide clad with Zircaloy-2. The reactor will operate at a normal pressure of 615 psia. Light water is used as the coolant and moderator and will be recirculated through the boiler region by pumps located external to the reactor vessel. Controlled recirculation will provide an additional means of controlling the output of the reactor. Steam generated at 489F in the boiler region will be passed through the superheater region raising the temperature to 825F for ultimate use in a condensing steam turbine. Present specifications for the Pathfinder reactor with nuclear superheater are listed in the table on the following page.

Various potential and hypothetical hazards and incidents have been reviewed and results of analysis are presented in this Safeguards Report. Safety is a prime consideration in the design of the Pathfinder Atomic Power Plant and its reactor and safeguards analyses will continue.

In this report, it is shown that the reactor building, which is designed to meet all applicable codes, is structurally stable and capable of withstanding internal pressures resulting from maximum release of steam or coolant. It is also shown that the reactor building is capable of containing radioactive fission products released into the building during any reasonable postulated incident. It is concluded that the Pathfinder Atomic Power Plant will not constitute a hazard to the health and safety of the public and operating personnel.

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DESIGN DATA

PATHFINDER ATOMIC POWER PLANT

PLANT

Power, boiler region.....157,200 kw
Power, superheater region.....42,400 kw
Reactor steam flow at rated power.....616,125 lbs/hr.
Total core power.....199,600 kw
Gross electrical capability.....66,000 kw
Net electrical output.....62,000 kw
Net efficiency.....31%
Steam outlet pressure.....535 psig
Reactor operating pressure.....600 psig
Temperature, boiler region.....489 F
Outlet temperature, superheater region.....825 F
Gross heat rate.....10,199 Btu/kwhr
Reactor building size.....50 ft dia. x 120 ft

REACTOR

Vessel size.....11.5 ft x 31.5 ft
Total core dimensions.....6 ft x 6 ft
Dimensions of superheater region.....6 ft x 30 in
Fuel, boiler region (Zr clad).....approx. 1.8% enriched UO₂
Fuel, superheater region (S.S. clad).....approx. 93% enriched UO₂
Fuel, loading, boiler region.....117 kg (U235)
Fuel, loading, superheater region.....42 kg (U235)
Average heat flux, boiler region.....128,000 Btu/hr/ft²
Average heat flux, superheater region.....77,800 Btu/hr/ft²
Maximum heat flux, boiler region.....462,000 Btu/hr/ft²
Maximum heat flux, superheater region.....245,000 Btu/hr/ft²
Recirculation rate.....60,000 gpm
Recirculation pump power.....794 kw
Neutron flux.....approx. 5×10^{13} n/cm² sec

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I.5	Typical Blister Formation	Battelle Memorial Institute
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I.7	Cermet Fuel Cell Model	A-C Dwg. 43-024-219
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Section I

REACTOR

1.1 General Description

The reactor complex which is discussed in detail in this Section consists of the reactor vessel and its internals, the recirculation pumps, valves, and piping, and the control rod drives. These items are described briefly in the following paragraphs and in more detail in the subsequent subsections.

a. Reactor and External Components (refer to Figure 3.1)

1. Reactor Pressure Vessel

Fabricated of stainless steel-clad carbon steel, the reactor vessel contains the nuclear fuel, associated support structure and internal components required to produce superheated steam under controlled conditions.

2. Recirculation Pumps

The three recirculation pumps force water at temperatures approaching the boiling point through the boiler fuel elements at a total rate of about 24×10^6 lbs./hr. (60,000 gpm). Of this flow about 600,000 lbs./hr. are boiled off and the remainder is returned to the suction of the pump to complete the hydraulic circuit.

3. Butterfly Valves

A corrosion resistant stainless steel butterfly valve is located in each 22 inch dia. pipe of the recirculation system.

long. The nuclear fuel produces saturated steam from the recirculation water. Water serves as both a coolant and a moderator.

2. Superheater Region

The superheater region of the reactor core resembles a tube-type heat exchanger consisting of 415 fuel bearing tubes approximately 1 inch in diameter. Saturated steam flows through the annular elements, is superheated to 825 °F, and exits at the bottom of the pressure vessel directly to the steam turbine.

3. Steam Separators

A total of 45 10-inch diameter centrifugal type steam separators are arranged around the outside of the boiler region in the downcomer space. They are designed to remove entrained steam from the recirculation water. Cavitation of the pumps is thereby prevented.

4. Steam Dryers

The centrifugal type steam dryers are located above the water-steam interface. Saturated steam generated in the boiler region is passed through the steam dryers to remove moisture particles which were entrained in the steam as it left the liquid interface. Use of steam dryers minimizes erosion damage to the superheater tubes and fuel.

5. Boiler Element Hold Down Structure

Although the weight of a boiler element is approximately equal to the upward force of the coolant pressure which tends to float

the element, sturdy hold down structure is employed above the boiler region to align the elements and prevent movement.

6. Boiler Baffle

The boiler region is completely enclosed by a water tight, circular baffle of stainless steel designed to position the boiler elements, prevent by-pass of coolant flow and serve as the mounting surface for the upper hold down structure.

7. Control Rods

Control of the reactor power level is maintained by manual adjustment of control rods located in both the boiler and superheater regions of the reactor.

8. Control Rod Drives

Rack and pinion type control rod drives are mounted on top of the pressure vessel and are employed to position the control rods in the nuclear core to effect a change in power level. The drives also force the control rods into the core in the case of an emergency shut down.

9. Internal Support Structures

Both the boiler and superheater fuel elements are supported by structural members of stainless steel. These members are of adequate rigidity to maintain position and alignment of the fuel elements with respect to each other and with respect to other components such as control rods and the hold down structure.

The steam separators and steam dryers are supported on stainless steel shelves by brackets welded to the wall of the pressure vessel. Mountings are designed to facilitate removal where necessary and prevent flow-induced vibrations in the components.

10. Feedwater Distribution Ring

The feedwater is returned to the vessel and injected into the recirculation water through a feedwater distribution ring located above the recirculation water outlet pipes. This insures a maximum amount of mixing with the recirculation water. Adequate subcooling of the recirculation water is thereby maintained and cavitation damage to the recirculation pumps is minimized.

1.1.1 Boiler Fuel

Fuel Element Design

The fuel element assembly consists of a bundle of 81 fuel rods in a 9 x 9 array, with a nozzle assembly on one end and a handling fitting on the other. The rod bundle has two regions containing rod sections of different diameters (see Boiler Data Sheet). Those in the upper half have a reduced diameter to compensate for reactivity loss due to increased steam content.

The fuel handling fixture is designed to accommodate a quick release pin type handling tool. The upper section of fuel rods is rigidly supported and spaced by means of a cast stainless steel grid fastened with screws to the fuel handling fixture.

The nozzle assembly incorporates a spring which absorbs overall difference in length due to tolerance and the differential thermal expansion between the

Zircaloy clad rods and the stainless steel shroud which supports the upper hold down grid. The nozzle assembly provides for a smooth transition from the circular flow channel in the nozzle to the square flow channel in the element. The fuel rod support grid for the lower rod bundle is fastened to the nozzle assembly with screws.

Each fuel rod is fastened to the support grid with a retaining ring capable of supporting an axial thrust load in excess of 100 pounds. The ring and groove design is such that vibration is avoided. The rings are made from PH 15-7 Mo stainless steel which is corrosion resistant and has the required mechanical properties.

The bundle of fuel rods is axially supported at both ends and laterally supported every 18 inches by the grid and the tube sheets. This type of construction allows the rods to axially expand separately within the confines of the rod bundle. The fuel rods making up the bundle are composed of four 18-1/2 inch fuel rod sections fastened together over tube sheets by a screwed joint. These fuel rod sections contain centerless ground UO₂ pellets clad with Zircaloy-2 tubing. End caps for the fuel rod sections are inert arc fusion welded to the tubing. Each section provides sufficient space to allow for the differential expansion between the pellet stack and the cladding as well as for the fission gases released during operation.

Material Properties and Technology

Zircaloy-2 which has a nominal composition of 1.5% Sn, 0.1% Fe, 0.1% Cr, 0.05% Ni, 0.035% C and the balance Zr, is used for the Pathfinder

reactor boiler fuel cladding because of its excellent nuclear characteristics and because of its compatibility with the boiling water environment. The minimum yield strength of Zircaloy-2 at the operation temperature of the boiler cladding (approximately 600 F) is 20,000 psi. The maximum design stress in the material is 11,000 psi, therefore, the material is adequately strong.

Information in the literature attributes a high degree of corrosion resistance to Zircaloy-2 in high purity water up to approximately 600F. Corrosion rates reported for Zircaloy-2, from tests under the conditions being considered, are comparable to those known for 304 stainless steel. In general, the corrosion rate for Zircaloy-2 is less than 0.1 mil per year in the pH range of 5-10. In EBWR it has been used successfully in 500F water at pH 7.

Zircaloy-2 as a fuel cladding material, in water reactors, has a good record of performance from the standpoint of corrosion resistance. PWR, EBWR, VBWR and Navy sponsored reactors have demonstrated the satisfactory performance of Zircaloy-2 in this respect.

There are some undesirable corrosion characteristics of Zircaloy-2 which have been considered. The mechanism of Zircaloy-2 corrosion is similar to stainless steel in that an oxide film forms on the metal surface which then protects it from further attack. This film can break down after a period of time, particularly at temperatures above 600F, and accelerated or "breakaway" corrosion occurs. Since the "breakaway" point may vary from lot to lot of Zircaloy-2, it has become common practice to specify some

standard accelerated corrosion test for each lot. One such standard test is to subject the metal to steam at 750F and 1500 psi for 24 hours. Metal which meets the requirements of the test is judged acceptable for reactor water environment.

Zircaloy is also susceptible to "hydriding" at temperatures above 850F which leads to embrittlement. Zircaloy-2 clad fuel elements which have failed have shown evidence of accelerated local corrosive attack accompanied by hydriding. Deliberate attempts to rupture fuel element cladding under irradiation by permitting water to penetrate the cladding through a drilled hole have been successful in one case*. The failure mechanism can be attributed to accelerated corrosion in the high temperature steam atmosphere accompanied by hydriding which embrittles the metal permitting rupture. Attempts to duplicate this type of failure in another case** have been unsuccessful.

Proper fabrication and inspection of Zircaloy-2 materials using good manufacturing quality control procedures and an accelerated corrosion test of each lot of material will minimize the possibility of having defective boiler fuel elements in the Pathfinder reactor. When the integrity of the material is maintained, hydriding is not expected to be a problem.

Fabrication technology of the Zircaloy materials has been developed under other reactor programs and is well known. Fuel element assemblies with a high

* "Experimental Program in the VBWR", D. H. Imhoff, W. H. Cook, R. T. Pennington and C. N. Spolaris. ANS Annual Meeting. 1959.

** Nucleonics Week, June 2, 1960, p. 1.

degree of integrity have been built utilizing Zircaloy-2. Zircaloy has been used in conjunction with UO_2 in a number of reactors and considerable operating and testing experience has been acquired. The Pathfinder boiler fuel and its operating environment are essentially the same as these applications.

Uranium dioxide has good irradiation stability and resists corrosion in case of cladding failures. It is readily available in usable forms and can be easily reprocessed. UO_2 pellets which meet Pathfinder requirements have been fabricated and used in large quantities.

The reactor boiler grid plate is fabricated on type 304 stainless steel. During reactor operation and during fuel handling operations there will be relative movement between the boiler fuel element nozzles and the grid plate. Armco 17-4 Ph in its hardened condition was selected as a nozzle material to minimize galling and seizing. This decision was based on the results of piston and cylinder wear tests run at 500 F and reported in the "Corrosion and Wear Handbook for Water-Cooled Reactors". A test is being run in 489F water to confirm published results. If further treatment is necessary the nozzles will be nitrided or hard chrome plated.

The Ph 15-7 Mo retaining rings used to fasten the fuel rods to their support grids has a high yield strength at operating conditions and in the Pathfinder environment its corrosion resistance is superior to stainless steel.

The large spring in the nozzle assembly is made of Inconel "X" and is expected to have negligible relaxation during its life under Pathfinder conditions. It will be operated at a stress level of about 54,000 psi which

is a conservative working stress at these temperatures (the material can be worked at stresses of 70,000 psi).

All other components of the boiler fuel assembly are fabricated of types 304 or 304L stainless steel. These materials are generally acceptable for reactor use.

Irradiation Experience

There has been a considerable amount of irradiation testing performed by others on fuel elements composed of UO_2 pellets clad in Zircaloy-2 tubes. The range of experiments conducted at Hanford, Chalk River and Bettis Atomic Power Department include the heat fluxes, surface temperatures, and burnups expected in Pathfinder. UO_2 pellets with O:U ratios less than 2.05 are stable under irradiation. Fission gas release, thermal conductivity, thermal expansion and certain other characteristics have been predicted based on these experiments. Except at temperatures higher than 1200 F, Zircaloy is compatible with UO_2 and no reaction is expected at the interface temperature found in Pathfinder.

Under irradiation the Zircaloy cladding experiences an increase in strength and a decrease in ductility but retains its pre-irradiation integrity.

The Shippingport and Chalk River reactors have operated successfully with UO_2 pellets, Zircaloy clad type fuel elements; the Dresden BWR is operating, and other reactors are scheduled to operate with this type fuel.

On this basis it is felt that additional irradiation testing is not required to prove the integrity of the boiler section fuel.

BOILER DATA

Core

Mean outside diameter	in.	69.0
Mean inside diameter	in.	31.3
Active fuel length	in.	72
Volume	ft ³	123
Number of fuel element assemblies		96
Average power density	KW/Ft ³ core volume	1278
Average specific power	KW/Kg of Uranium	23.8
Average heat flux	Btu/hr/ft ²	128,000
Maximum heat flux	Btu/hr/ft ²	462,000
Estimated burnout heat flux	Btu/hr/ft ²	1,000,000
Maximum fuel temperature	°F	3850
Maximum surface cladding temp.	°F	514
Inlet cooling velocity	Ft/sec	13.9
Exit liquid coolant velocity	Ft/sec	19
Exit steam velocity	Ft/sec	32
Exit voids	%	38
Pressure drop	Psi	18
Fuel burnup	MWD/T	10,800
Core lifetime	Yr.	1.4
Fuel enrichment U-235	w/o	1.8
Fuel loading, U	Kg	6560

Fuel Element Assemblies

Bundle Size	in.	4.735	° Rm. Temp.
Number fuel rods/bundle		61	
Rod sections/rod		4	
Pellet diameter @ Room Temp.	in.	Lower Rod Sects. .348	Upper Rod Sects. .310
Clad thickness	in.	.028	.026
Rod diameter @ Room Temp.	in.	.400	.367
Box material		Zircaloy-2	
Clad material		Zircaloy-2	
Fuel material		UO ₂	
Fuel density	gm/cc	10.41	
Nozzle material		17-4 Ph	
Thermal power generated	MW	157.2	
Thermal power to coolant	MW	159.9	
Heat transfer area	Ft ²	4204	
Coolant flow area, lower half	Ft ²	9.6	
Coolant flow area, upper half	Ft ²	11.0	

Test Program

A. Vibration. Tests have been conducted to establish the nature of vibration experienced by a fuel rod under two phase flow. Single rod tests were conducted which simulated reasonably well the hydraulic diameter, rod size, rod weight per inch, and rod end supports specified for Pathfinder. Fuel rod vibrations were detected by foil strain gauges and recorded on brush oscillographs under flow conditions (relative to velocity and void fractions) at and above those expected in Pathfinder. Results indicate that the vibrations will not be excessive with the 18-1/2 inch support spacing used on the reference element.

B. Fuel Element Box. A test was conducted in order to establish the reliability of the analytical method of determining the critical stresses in the corners of the fuel element box. An aluminum box of Pathfinder dimensions was equipped with strain gages and subjected to various internal pressures. Experimental results and the predicted values agreed within 8%.

C. Nozzle Galling. A test fixture has been fabricated to determine the galling characteristics of various nozzle materials. The parts used in the test are of the same geometry and will experience the same temperature and pressure conditions as the nozzle in Pathfinder. Testing is now underway.

D. Prototype Boiler Fuel Element Assembly. A prototype fuel element assembly similar to the design shown in Figure 1.2 has been ordered for the purpose of optimizing tolerances, and evaluating fastening schemes and handling ease. It will also be used for pressure drop and flow tests. This prototype utilizes stainless steel rods in place of Zircaloy clad fuel rods.

E. Fabrication Development. A total of twelve fuel rods were ordered from each of four potential fabricators of the core. Except for the enrichment of the uranium dioxide, these qualification rods will be fabricated with the same geometry, materials, and requirements as are specified for the Pathfinder core.

These rods will serve as a qualification of the potential vendors' capabilities for fabricating fuel rods of the quality required for Pathfinder. The completed rods will also be given dimensional, metallurgical and other tests.

Major difficulties in fabrication if any exist, should be discovered. This will permit appropriate changes in design prior to fabrication of the full loading for the boiler.

F. Heat Transfer Tests. Three series of heat transfer tests will be conducted to experimentally determine the performance of the boiler fuel element. The first of these will be a series of "burnout" tests, which will be performed on electrically heated single-pin test sections. Sections to be tested will have hydraulic diameters equivalent to those found in the upper and lower fuel element halves respectively. These tests will show that burnout will not be experienced at heat fluxes and steam qualities 50% greater than those anticipated in the Pathfinder reactor.

This next series of tests will be performed on a test assembly which provides a mock-up of the boiler. The test section is shown in Figure 1.5.

The water enters the test section on the right where 10 pins are used to mock up the 81-pin region. The middle bundle of 8 pins, mocks up a 64-pin region, and the left bundle of 6 pins mocks up a 49-pin region. Although the core design no longer includes a 64 or 49 pin region, the test section will still satisfy the original requirements of evaluating the effect of forced circulation on the void fraction and pressure drop.

It is expected that extrapolation can be made from these tests to obtain reasonably accurate design data for Pathfinder.

The last series of tests will be performed on a full size, mockup element, to obtain accurate flow, void and pressure drop data at exact reactor pressure and temperature conditions.

G. Fluid Flow Tests. Much useful steam void and pressure drop data will be obtained from the heat transfer tests. From those tests good full and part load hot spot or hot channel data will be obtained. To supplement the heat transfer data, tests will be performed in a low pressure loop, which uses air to simulate the gaseous phase. This test loop will be used to provide information on void location and pressure drop as affected by variations of flow, voids and spacing of element components. Since the air-water data is not directly applicable to steam-water pressure drops, the sections tested in the heat transfer loop will also be tested in the air-water loops to give better comparisons.

1.1.2 Superheater Fuel

Fuel Element Design

The superheater fuel element assembly (Figure 1.4) consists of two concentric fuel bearing tubes, a central burnable poison rod, spacers, and end fittings. Each fuel tube consists of a 0.015 inches of UO_2 -stainless steel cermet sandwiched between two 0.006 inch stainless steel cladding tubes. Type 316L stainless steel is used throughout the assembly.

Specific dimensions of the superheater fuel are given in the data sheet and Figure 1.4.

The cermet is made from nearly spherical UO_2 powder (44 to 53 micron diameter) mixed with type 316L stainless steel. The concentration of UO_2 in the cermet is 20.4 w/o or 15.9 v/o.

The poison rod consists of a 0.449 inch diameter compact containing approximately 0.1 w/o boron carbide in alumina at about 70% theoretical density. It is clad with 0.010 inches and stainless steel. The mechanical design of the element is such that the burnable poison in the center rod is contained and retained in a manner that eliminates any possibility of losing it suddenly.

The fuel tubes and the poison rod are fastened together by a spacer and support fitting at the lower end. This spacer, plus three full length straight spacer wires attached to the outside diameter of the poison rod and each fuel tube, maintain concentricity of the

components. The fuel tubes are not axially restrained at the upper end and are free to expand in that direction. The steam flow channels are sized to assure proper coolant flow distribution for optimum heat transfer.

The superheater fuel assembly is about 6-1/2 feet long with an active fuel section of 6 feet long.

Materials Properties and Technology

The corrosion of type 316 stainless steel, and austenitic stainless steels in general, has been investigated. These materials are now commonly used in superheaters delivering steam at 1150 to 1200 F. Their use for 825 F superheated steam is expected to present no problem. Corrosion rates for austenitic stainless steel in 1100 F steam is reported to be 0.02 mils per 12,000 hours. ⁽¹⁾ Experimental superheater coils have been tested with metal temperatures exceeding 1350 F (steam 1250 F) and steam side corrosion rates are of the order of 1 to 2 mils per year. ⁽²⁾ Although little information exists on the effect of irradiation on steam corrosion rates, there is no significant effect on corrosion rates for water systems. ⁽³⁾ Several brief tests in a superheated steam in-pile loop have not given any indication of accelerated general corrosion or fretting type corrosion at points of mechanical contact.

(1) Transactions of the ASME, Vol. 66, page 277

(2) Transactions of the ASME, Vol. 79, page 307

(3) DePaul, D. J., "Corrosion and Wear Handbook for Water Cooled Reactors", p. 293
TID-7006, March 1957

Erosion due to impingement of wet steam will be minimized by keeping the moisture content of the steam low, of the order of 0.1%, and by providing adequate lengths of heavier stainless steel entry sections above the fuel bearing portions of the fuel tubes. Erosion effects are expected to be significantly less than those due to corrosion.

The cermet type fuels have been under development for many years and have been fabricated into fuel assemblies. Uranium dioxide - stainless steel cermets clad with stainless steel have been or are being fabricated for use in the Army Package Power Reactors, Borax V, Spert III., VBWR, and the Gas Cooled Reactor Experiment as well as for several classified projects. The clad cermet fuel under consideration has good physical properties (essentially equal to those of stainless steel). Very thin cladding can be metallurgically bonded to the fuel to give a structurally sound tube with excellent heat transfer characteristics. Quality control is maintained during preparation of matrix materials and other steps of the fabrication processes. The ideal cermet would consist of spherical fuel particles arranged in a geometrical array thereby permitting a maximum quantity of high strength metal to surround each fuel particle. In practice, the ideal condition cannot quite be attained. The optimum distribution achieved consists of a random orientation of near spherical particles with the majority separated by an average amount of metal.

Fragmentation, stringering, and/or agglomeration of fuel particles in the matrix must be nearly eliminated as these conditions can lead to

general swelling and blister type failure. Fracture of individual oxide particles allows higher percentages of fission product gases to be released. Stringering and agglomeration cause planes of weakness by decreasing the relative metal volume in local areas. The cumulative effect is a higher susceptibility to radiation damage.

The core-to-clad interface must be made metallurgically bonded to prevent accumulation of fission product gases and resultant blister formation. A good bond minimizes general swelling by increasing the effective strength of the element.

The matrix and cladding of the finished fuel element should have maximum ductility and strength as required for containment of cermet swelling and fission product gases, and to minimize fuel element failure.

Two recent surveys present an excellent coverage of dispersion fuel element fabrication processes, and the structures obtained. (4 & 5) All established methods utilize some form of powder metallurgy; viz. cold or hot pressing, swaging, extrusion or rolling. The cermet is clad with a protective sheath by hot-working, cold-working followed by a diffusion heat treatment, or by an integral forming operation of the core and clad.

- (4) Weber, C. D. "Progress on Dispersion Elements", Progress in Nuclear Energy, Series V, Volume 2, "Metallurgy and Fuels", Pergamon Press, New York, 1959, pp. 259-352 (Uncl.)
- (5) Cunningham, J. E., R. J. Beaver, and R. C. Waugh, "Fuel Dispersions in Stainless Steel Components for Power Reactors", Paris Conference on Fuel Element Technology, TID-7546, Book 1m USAEC, Technical Information Service Extension, March, 1958, pp. 243-268, (Uncl.).

The mechanism of irradiation stability of UO_2 -stainless steel dispersion fuel elements has been studied during the last ten years. Two reasonable hypotheses of the mechanism of failure have been developed from theoretical considerations and in-pile test results. While these mechanisms differ in approach, they provide comparable estimates of fuel life and stability.

A. BMI Program

A study at Battelle Memorial Institute has developed a method of correlating irradiation stability with cermet structural properties using a simple failure-by-creep mechanism. (6 & 7) Irradiation stability of failure is affected by material properties of the matrix, structure of the cermet, irradiation temperature, fuel burn-up, and irradiation duration.

The creep mechanism analytical approach uses failure processes experimentally observed in UO_2 stainless steel fuels. (6 & 7) In reactor environments, fission product gases are formed which collect in voids around UO_2 -particles and in the matrix, and in pores within the fuel particles. As irradiation continues, gas pressure builds up in these areas resulting in a stressing of the matrix metal. This force causes creep deformation of the stainless steel matrix, and general swelling of the fuel element. As gas pressure continues to increase, a point is reached where the creep strength

(6) Keller, D. L., L. E. Hubert, and B. W. Dunnington, "A Method of Correlating Irradiation Effects in Dispersion Fuels", Battelle Memorial Institute, BMI-1408.

(7) Keller, D. L., Private Communication. Battelle Memorial Institute

is exceeded, and localized fissures occur in the matrix between adjacent voids or fuel particle spaces. The gases follow fissure formations to the cermet edge where pressure against the cladding forms blisters on the surface of the fuel element. Additional gas pressure causes the blister to expand until it bursts, releasing gases to the coolant steam. Figure 1.5 shows the resultant blister formation on the surface of a fuel plate. Study of in-pile specimens at BMI has shown that fission gas blisters rise to a maximum of 0.015 inch on the surface, and occasionally burst, usually in three radial cracks developing from the center of the blister.⁽⁷⁾

Using this experimentally observed stability pattern, a creep mechanism formula has been developed for evaluating burn-up and fuel life for UO_2 -stainless steel cermet elements. The complete derivation and analysis of the formula is given by Keller, Hulbert and Dunnington.

Figure 1.6, Curves A through D, are plots of predicted performance derived from experimental data using above theory. Data points are included and show the correlation with theoretical curves rather well. The analysis covers the temperature and burn-up ranges experimentally investigated: 600-1840 F.

It is important to note that the values obtained in this study apply only to fuel cermets with the specific fuel matrix ratios as fabricated by BMI. However, by inserting the design variables of burn-up, fuel life

(7) Keller, D. L., Private Communication. Battelle Memorial Institute

duration, maximum temperature of the fuel, etc., this formula may be used to approximate the life and stability of Pathfinder superheater fuel elements as discussed under heading Irradiation Experience.

B. KAPL Program

A similar study of UO_2 -stainless steel irradiation stability is being made at Knolls Atomic Power Laboratory. This study also compares the effect of fabrication and irradiation variables with in-pile test results. Its purpose is to establish the functional limits of the dispersion fuel system, and to develop an optimum cermet fuel element.⁽⁴⁾

The basic model for design calculations is an individual fuel cell consisting of one fuel particle, its fission product damaged matrix zone, and the surrounding metal matrix separating other fuel particles as shown in Figure 1.7. Each fuel cell is treated as an individual fuel element with the matrix metal acting as a pressure vessel restraining the fuel particle stresses and fission gas pressure. Several unit cells, arrayed in a hexagonal geometry composes the cermet dispersion structure. With this model, the irradiation stability variables of volume percent fuel, fuel particle size, fuel particle density, fuel element temperature, fuel burn-up, and matrix strength and ductility are analyzed simultaneously. Comparison of these design variables with irradiation test results has shown the practicality of this method.

(4) Weber, C. D., "Progress on Dispersion Elements", Progress in Nuclear Energy, Series V, Volume 2, "Metallurgy and Fuels", Pergamon Press, New York, 1959, pp. 259-362 (Uncl.)

The effect of each of the major irradiation stability variables is discussed briefly below:

a) Volume Percent Fuel. The higher the volume of fuel, the lower the metal spacing between particles of fuel, and the lower the resultant strength of the cermet. UO_2 -stainless steel cermets in the range of 10 to 30 w/o oxide, (8-24 v/o oxide), have been investigated.

b) Fuel Particle Size. For a given fuel volume, the smaller the particle size, the larger the relative fission product damage zone, and the lower the effective matrix strength. KAPL has worked primarily with 50-200 micron diameter fuel particles. Below 50 microns, fission product damage zones tend to overlap. Larger size particles, greater than 200 microns, may be used effectively in elements having greater cermet thicknesses. However, in order to achieve orientation of the fuel particles, prevent cracking and stringing, etc., the fuel particle size should be small in relation to cermet thickness.

c) Fuel Particle Density. Limited porosity in a fuel particle is attractive as a space for fission formed gases to collect and will tend to prevent gross swelling of the fuel element. These KAPL studies have shown the value of this porosity, but the porosity required for a given fission gas evolution has not yet been determined.

d) Fuel Element Temperature. For a given quantity of released fission product gases within a fuel element, higher element temperatures produce higher gas pressures and higher strain and swelling effects. Moreover the strength of the element decreases at elevated temperature.

e) Fuel Burn-up. A fraction of the fission products recoil from the fuel into the surrounding matrix reducing its strength and ductility in the recoil range. As burn-up proceeds, a volume increase in the fuel particle strains the metal immediately surrounding the particle. This mechanical stress, along with stresses from fission gas pressure strain the metal and as a result micro-fissures, cracks, and swelling occur in the matrix and element structure.

f) Matrix Strength and Ductility. The ductility of stainless steel depends on the strain rate in biaxial tension.

Fabrication Processes

At least three fabrication processes exist which could be used to produce a cermet fuel element for the Pathfinder superheater.

a) Hot-Roll Cladding of Cold-Pressed and Sintered Core Compacts.

This process has been used by ORNL, Sylcor, M & C Nuclear, Martin and Battelle Memorial Institute to manufacture cermet plates and concentric tubes. Core compacts containing 20 to 40 w/o oxide have been prepared by weighing, blending, and cold-pressing component powders into rectangular shapes. Frames and cladding are welded around the compact, and the composite is hot-rolled to develop the core-to-clad bond and to densify the core meat. Subsequent cold-reductions with intermediate heat treatments are used to obtain the final size of the element. While considerable manufacturing experience exists with this method, homogeneity, fragmentation, and stringering - agglomeration characteristics are not easy to control.

b) Cold-Work and Diffusional Heat Treatment Cladding of Cold-Formed Cores.

In this process, the cermet core (made from weighed and blended component powders) is cold-formed by pressing, rolling, or swaging. Then frames and cladding are positioned around the core, and the composite is cold-worked. By the use of alternate diffusion heat treatments and cold-working operations, the core is densified and the core-to-clad bond is developed. The Martin Company has utilized this process to prepare tubular fuel elements of UO_2 -stainless steel. In their process, the blended and mixed powders are cold-formed into a cermet strip by rolling. This method uses a minimum of cold reduction to form a green compact of approximate finished size. The green compact is sintered and cold-rolled to densify it. The high density strip is formed into a fuel tube, and cladding tubes are cold-drawn over it. Subsequent diffusional heat treatments and cold-draw sizing give a fully dense cermet and develop a good core-to-clad bond. The method has been used to fabricate one complete core loading for reactor critical experiments and a core loading for the PM-1 reactor.

c) Integral Core-Clad Forming of Cold Binder Cores By Hot-Working

In this process an organic binder is utilized to improve fuel distribution and agglomeration. Hot-working and cladding of the core compact reduces stringering and fragmentation of the fuel particles. While this process has produced the best UO_2 -stainless steel cermet structures reported, it has not been adequately developed for commercial production.

The best commercial process available appears to be process b) above.

The following is a communication from Martin dated May 18, 1960 discussing their experience.

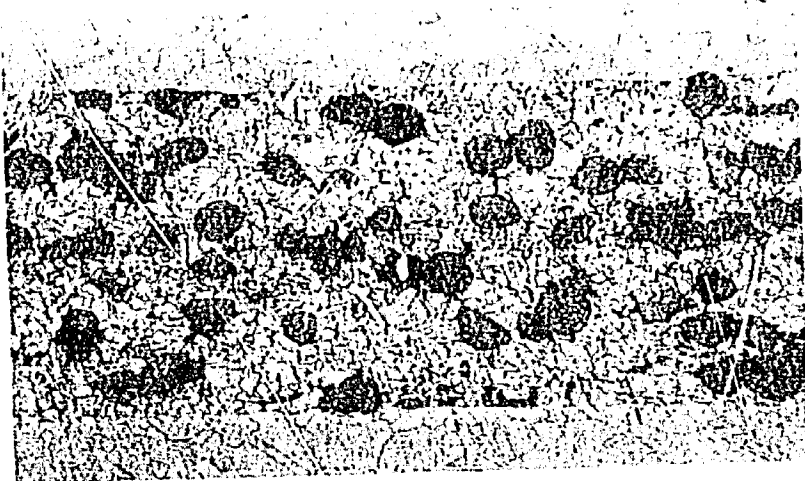
"We are pleased to submit the following remarks regarding our experience with the development and subsequent production of stainless steel clad, stainless steel uranium dioxide cermet tubular fuel elements.

In 1954, The Martin Company decided to design and build portable packaged nuclear power stations. The requirements placed a premium on getting the most power out of a small package. Bulkiness and weight, were of necessity, held to a minimum. The highly enriched core of the reactor was analyzed and designed for optimum efficiency. The core specifications called for the largest heat transfer area, minimum of extraneous supports, structural integrity, and lightest weight. Various methods of fabricating fuel elements to best satisfy these requirements were analyzed and our decision was to fabricate fuel elements in a tubular shape. To assure maximum properties in the core, a new process was developed. In this new process, the finely powdered stainless steel-uranium dioxide mixture is passed through a special rolling mill and formed into a ribbon of near-final thickness before being enclosed in its stainless steel cladding. Subsequent fabrication to finished dimensions increases the length of the fuel core by only 1/2 compared with elongation ratios of approximately 7 to 1 in conventional practice.

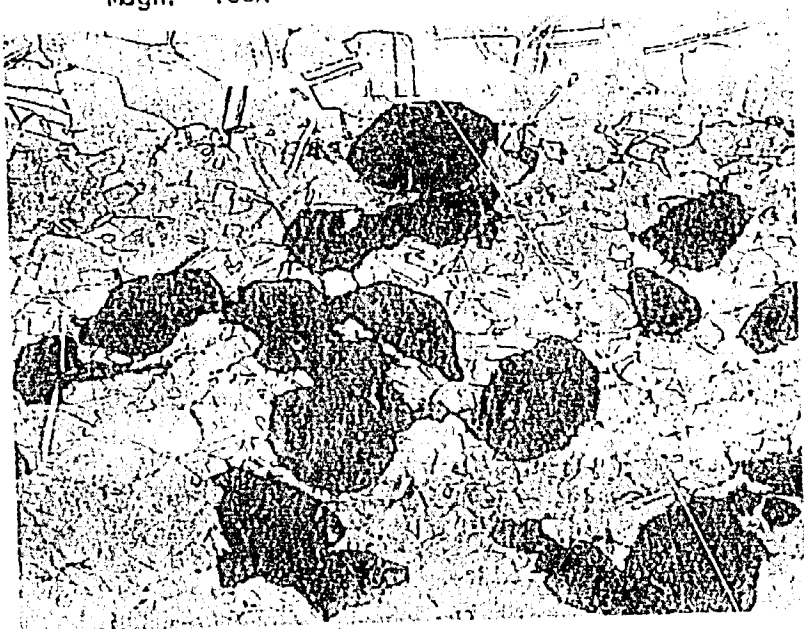
The final product is made to size with virtually no oxide break-up or stringering and with excellent control of fuel location. These conditions have been consistently reproduced in production elements. Sample fuel elements were irradiated in the high flux hole of the Materials Test Reactor, Arco, Idaho and the nuclear characteristics of the design were demonstrated by the fact that they achieved a 42 per cent burn-up with no visible changes in the element. In addition, thermal and mechanical shock tests were conducted in conjunction with metallographic tests to determine the nature and strength of the bond between core and claddings. Static and corrosion tests were performed under temperatures and pressure conditions more severe than those encountered at the MTR installation. Results in all cases were above expectation.

The attached photomicrographs clearly show the uniform dispersal of uranium dioxide grains throughout the core with minimum break-up and stringering. They also show the close crystalline bonding between cladding and core achieved in this process.

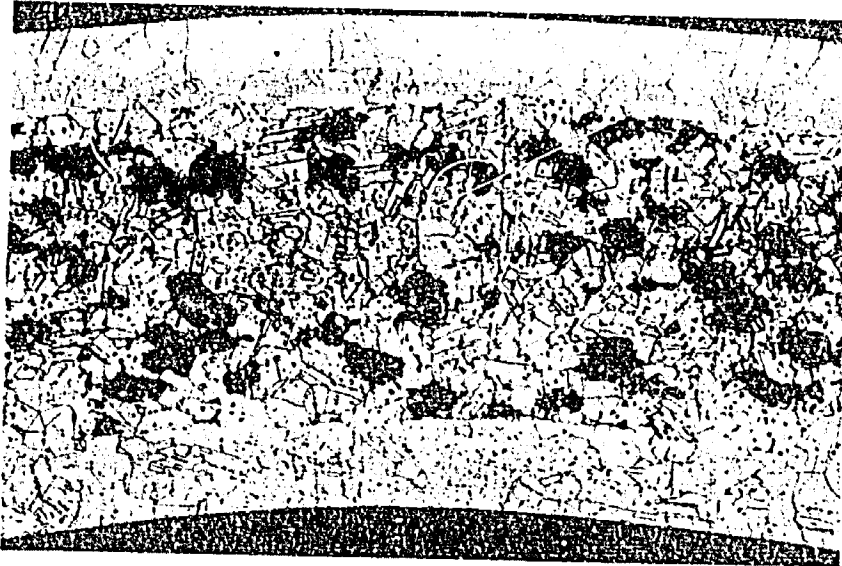
(continued on page 1.26)



FLAT PLATE ELEMENT LONGITUDINAL
Etchant: Mixed Acids
Magn: 100X



FLAT PLATE ELEMENT LONGITUDINAL
Etchant: Mixed Acids Magn: 250X
30 w/o UO₂

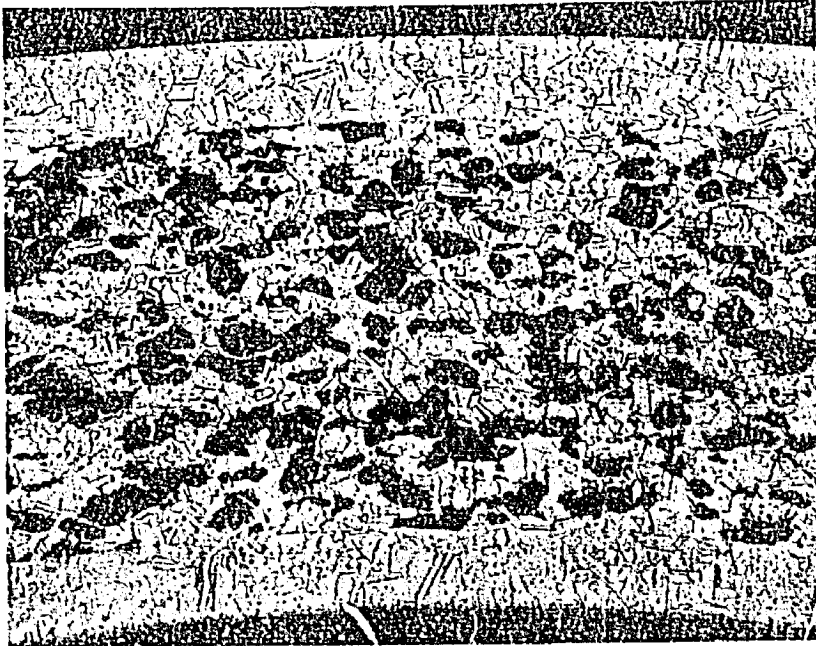


MFR TRANSVERSE SECTION
Etchant: Vilella's
Magn: 125X

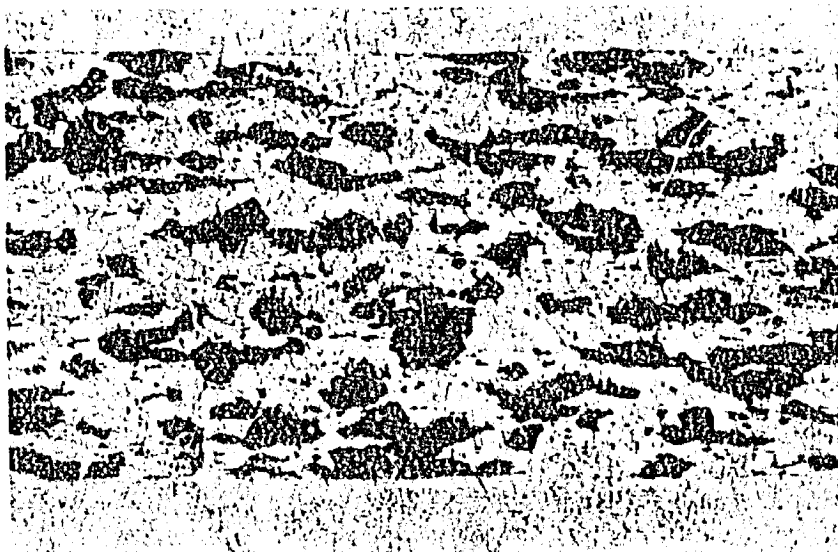


MFR LONGITUDINAL SECTION
Etchant: Vilella's
Magn: 125X
25 w/o UO₂

(1.25b)



FM-1 TRANSVERSE SECTION
Etchant: Vilella's Magn: 75X



FM-1 LONGITUDINAL SECTION
Etchant: Vilella's Magn: 75X
30 w/o UO_2

Specific test data on tubular fuel elements developed in 1956 are available if you desire to see it. A full scale core recently produced for the PM-1 air transportable reactor is presently undergoing critical tests and data should be available in several months.

Martin is continuing its efforts to improve the technique of producing production tubular elements by powder rolling and we firmly believe that as our experience increases, even better performance can be expected over what we now consider high performance fuel elements."

Irradiation Experience

A uranium dioxide stainless steel cermet clad with stainless steel was selected as the reference superheater fuel for Pathfinder because sufficient fabrication techniques had been established and sufficient irradiation data existed to proceed with the design of an element and to warrant its being placed in the reactor without the need for a major fuel development program. This approach is still considered to be valid even though in recent years similar information has been accumulating on other types of fuel adaptable to superheater use. The irradiation data referred to above is tabulated primarily in CF-58-2-71, "Summary of UO₂-Stainless Steel Dispersion Element Irradiation Experiments," by Haynes, Neill & Schaffer of Oak Ridge National Laboratory. More recently, cermet type fuels very similar to that specified for Pathfinder have been irradiated at temperatures slightly higher than the peak calculated Pathfinder fuel temperatures. These data were obtained as part of the Gas Cooled Reactor Experiment program and are tabulated in the following table.

GCPE-1 Cermet Fuel Element Irradiation Test Results*

Sample Number	UO ₂ w/o	Exposure Time Days	U-235 Burnup a/o	Condition of Sample After Test	Effective Temperature F
23	25	32.7	5.4	Good	1725
33	25	47.5	12.7	Good	1430
35	25	47.5	16.6	Good	1550
39	25	85.2	20.4	Large Blister	1400
40	25	85.2	20.7	Small Blister	1417
42	25	52.7	2.2	Good	1650
43	25	52.7	2.6	Good	1650
21	30	32.7	4.9	Small Blister	1655
31	30	47.5	17.5	Small Blister	1533
32	30	47.5	15.6	Ruptured	1395
36	30	85.2	25.1	Ruptured	1475
37	30	85.2	22.2	Ruptured	1300
44	30	52.7	2.4	Good	1650
45	30	52.7	2.8	Good	1650
22	30	32.7	4.9	Large Blister	1655 (1)
24	25	32.7	5.4	Good	1725 (1)

* From BMI-1408 and Personal Communications with BMI.

(1) Temperatures are estimates since thermocouples were inoperative.

GCRC-1 Cermet Fuel Element Irradiation Test Results*

Sample Number	UO ₂ w/o	Exposure Time Days	U-235 Burnup a/o	Condition of Sample After Test	Effective Temperature F
23	25	32.7	5.4	Good	1725
33	25	47.5	12.7	Good	1430
35	25	47.5	16.6	Good	1550
39	25	85.2	20.4	Large Blister	1400
40	25	85.2	20.7	Small Blister	1417
42	25	52.7	2.2	Good	1650
4	25	52.7	2.6	Good	1650
21	30	32.7	4.9	Small Blister	1655
31	30	47.5	17.5	Small Blister	1533
32	30	47.5	15.6	Ruptured	1395
33	30	85.2	25.1	Ruptured	1475
37	30	65.2	22.2	Ruptured	1300
44	30	52.7	2.4	Good	1650
45	30	52.7	2.8	Good	1650
22	30	32.7	4.9	Large Blister	1655 (1)
24	25	32.7	5.4	Good	1725 (1)

* From BMI-1408 and Personal Communications with BMI.

(1) Temperatures are estimates since thermocouples were inoperative.

(1.27)

It should be noted that these data are for 25 and 30 w/o UO_2 specimens whereas Pathfinder fuel is specified at about 20 w/o and that the irradiation temperature is, in most cases, above the approximate 1400 F hot spot temperature estimated for Pathfinder.

As discussed earlier, these data have been correlated by means of a theoretical model by Battelle Memorial Institute. The results of this correlation are shown as the solid line stability curves A through D in Figure 1.6. The actual data points are also shown in this figure.

It is logical to assume that for the same irradiation times, similar stability curves for 20 w/o UO_2 cermet fuel fabricated by the improved cold powder rolling process (as compared to the hot rolled fabrication of the test specimens) would fall above and to the right of those for the test specimens. Both the reduced fuel loading and the better fabrication technique lead to a greater average amount of stainless steel between fuel particles and a stronger fuel cermet less susceptible to irradiation damage.

In regard to Pathfinder fuel elements, BMI investigators estimated that a fuel burn-up in excess of 20 a/o U-235 in 5000 hours operation could be achieved at 1400 F with GCRE-1 quality fuel. With a better cermet structure, which is known to be obtainable, a burn-up of 30 a/o or greater can be expected according to BMI.

To be quite conservative, the estimated Pathfinder fuel stability curves E and F in Figure 1.6 are not plotted as far to the right as they might be. Calculated Pathfinder superheater fuel burn-up versus temperature Curves G, H and I (along the length of an element) is also plotted in

Figure 1.6 for comparison. The method of construction of these curves is illustrated in Figure 1.8.

In Figure 1.8, the scale in the lower left quadrant represents the distance traversed by the steam through the superheater core. In the upper left quadrant are curves representing axial burn-up distribution for the conditions indicated. (These curves are proportional to the respective neutron flux distribution). In the lower right quadrant are curves representing the fuel surface temperatures for the same conditions. Burn-up versus surface temperature curves are constructed in the upper right quadrant. The ordinate of these curves is obtained by proceeding up from the superheater fuel scale to the axial burn-up distribution curve for a particular condition. Similarly, the abscissa is found by proceeding to the right from the same point on the superheater fuel scale to the temperature curve. Typical construction lines are shown for two points on the 825 F, 9 month, hot channel curve. Also plotted in the upper right quadrant are lines A and B and point C which represent estimated limits of performance based on the results of irradiation tests at various Government laboratories. Successful operation is expected in the region below and to the left of these lines. Line A has been established as an "essentially 100% chance-of-success" of burn-up obtainable at various temperatures.⁽¹⁾ Variables that led to failure in some test samples are now recognized as controllable in the manufacture of fuel elements. Curve B is therefore considered to be

obtainable with present fabrication techniques. This conclusion is based on communications with AEC laboratory personnel. Point C is considered to be probably obtainable with optimum fabrication techniques.

From these curves it can be estimated that the Pathfinder superheater fuel will have a life of at least 6 (full power) months with a good possibility of achieving the desired life of 9 months.

The results of the GCRE irradiation tests also demonstrate that:

(1) Where failure has occurred it has been in the form of cracks or small blisters in the cladding. Blisters have been less than 0.015 inches in height and eventually relieve fission gas build-up beneath them through small cracks in the face of the blister.

(2) In blister failures, a crack is found in the cermet in planes of structural weakness. Failure is generally found in areas where the metal matrix has been depleted by fuel stringering or agglomeration.

(3) Fission gas release data indicate that gaseous fission products do not otherwise readily diffuse through the fuel clad.

Typical blister and crack type failures are shown in Figure 1.5. It is emphasized that the approximately 0.015 inch maximum height of these blisters is such that they cannot bridge across the steam flow passages.

In the Pathfinder reactor, the possibility of irradiation damage or mechanical failure that would result in channel blocking and flow stoppage, during normal operation, is considered to be non-existent.

It is expected that if any failures occur in Pathfinder superheater fuel they will occur individually in a random manner toward the end of the design life. Only a relatively small fraction of the fuel is exposed to the coolant in these failures. There are no corrosive or other chemical reactions anticipated that could result in a large volume increase and cause flow stoppage in a channel.

Based on the above discussion, it is concluded that:

1. Sufficient irradiation experience exists to safely specify cermet type fuel for Pathfinder and further test it under actual operating conditions.
2. The reactor can be operated without expectations of catastrophic failure of the fuel.

As part of the post-construction R & D program a progressive burn-up schedule for superheater cores is planned. Within this program whole cores or selected elements will be removed at various stages of burn-up and examined to determine their suitability for higher burn-up.

The safeguards considerations pertaining to the use of cermet superheater fuel in Pathfinder are discussed in Subsection 7.3.7.

Test Program

A. Thermal Shock Testing and Analysis of the Superheater Fuel Elements

During various shutdowns of the Pathfinder Plant, the superheater will be flooded with water at the then existing saturation temperature and pressure. The flooding water will prevent steam flow through the superheater, causing the temperature in the fuel elements to increase to some equilibrium value,

which is a function of the fractional power level at that time and the losses to the system. The incoming water will effectively quench the warmer fuel element, which will produce thermal stresses of varying degrees. In order to determine the effects of flooding, an analytical and experimental program was initiated. The program is intended to provide a good understanding of the conditions existing during flooding and to determine limitations, if any, on the superheater shutdown procedures.

In the analysis a conservative case was considered with a fuel element at 1400 F cooled by 400F water. Assuming an extremely conservative heat transfer coefficient, a fatigue life was computed to be 1000 shut-down cycles. Loss of ductility due to sigma formation in the fuel element and irradiation damage is insufficient to greatly decrease the number of cycles during the 9-month life of the fuel element. Analytical treatment of the controlling parameters indicated that the flooding water distribution within the element was critical. The flow pattern through the thin annuli of the fuel element was impossible to predict, due to the turbulence caused by formation of steam. A test is being performed in order to obtain empirical data over a range of cladding to water temperature differences. The fuel element geometry and flooding method are designed to reproduce Pathfinder conditions as closely as possible. Analytical results show that the heat capacity and generation of the fuel element has a negligible effect on the stresses and strains produced when the water contacts the cladding surface.

B. After-glow Test

The heat generation of a superheater fuel element after shut down is a function of the initial power level and the elapsed time from rod insertion. As previously stated, when the superheater is flooded, steam flow is stopped, which causes the fuel element temperatures to increase to some equilibrium value determined by the shut down heat generation, or, "after-glow", and the conduction and radiation losses to the superheater structure and moderator.

For use in design, the fuel element surface temperatures were calculated as a function of fractional superheater power (shown in Figure 1.9). In order to confirm these analytical results, a test is being performed, which utilizes a resistance-heated dummy fuel element located within a mock-up of a one-tube superheater structure, and surrounded by 600 psi saturated water. Various fractional power levels are being reproduced and the equilibrium surface temperatures are measured with thermocouples. Preliminary results show the calculated curve is conservative.

C. Fluid Flow and Heat Transfer Tests

A loop has been added to the boiling heat transfer loop to test superheater fuel elements. A schematic of the complete loop is shown in Figure 1.10. Both heat transfer and pressure drop data will be taken.

The tests will be run with an annular superheater test section. The two tubes fit together to form an annular steam passage. The two tubes are electrically insulated from each other and can be heated separately:

- a) Heating by both walls.
- b) Heating by outer tube only.
- c) Heating by inner tube only.

With this arrangement the three flow channels, of the superheater element can be simulated even though the spacing is not exact. This arrangement supplies a high degree of flexibility in determining the effect on the heat transfer coefficient of heating on both walls of the annulus.

The range of variables that the superheater test section is capable of covering is:

Reynolds No. 10,000 - 120,000

Heat Flux 40,000 - 300,000 Btu/hr-ft²

Maximum: Surface Temperature 1300 - 1400 °F

Mass Flow 50,000 - 600,000 #/hr-ft²

A small air flow test rig will be used at low temperature to obtain pressure drop data for the actual superheater geometry. Since the acceleration pressure drop associated with heating in the test section can be calculated, this test will extend the data obtained from the heat transfer loop.

D. Fabrication Development Program

The possibilities of high hot-spot surface temperatures, and desired high burn-up in Pathfinder dictates superheater fuel element manufacture to close tolerances. A fabrication development program was initiated to extend the present fabrication techniques to suit Pathfinder requirements.

The program has four main parts: cermet forming, fuel tube fabrication, fuel element assembly, and non-destructive testing. The objective is to refine present techniques to produce the desired fuel homogeneity, fuel particle uniformity, fuel dispersion and cermet density, with a minimum of

stringering in the fuel cermet. Bond integrity, dimensional tolerances, and reduction of stringering will receive emphasis in tube fabrication. Various assembly techniques will be used in order to select the one(s) capable of producing the required flow channel uniformity. From this program, a fabrication sequence is expected which can be used to produce superheater fuel elements meeting Pathfinder requirements.

Highlights of Specification for Superheater Fuel Element Assembly

The following is a very brief digest of the specifications used for procurement of the superheater fuel:

The fuel material shall be high fired uranium dioxide powder in spherical shape, with an 86.5% minimum uranium assay. The stainless steel powder used in the cermet dispersion shall be AISI Type #316L prealloyed stainless steel, spherical in shape as prepared by atomization techniques. The method of fabricating the fuel dispersion shall be left to the discretion of the vendor subject to the following limitations:

1. The cermet shall be fabricated so as to obtain a segregation and stringer free dispersion.
2. The quantity of uranium dioxide shall be 20.40 ± 0.50 weight percent with a cermet density of 95% of theoretical.
3. The fuel shall be dispersed in the matrix to $\pm 5\%$ of the mean average distribution.

The cladding shall be AISI Type 316L or 318 seamless tubing subject to ASTM specification #A-269. No defect deeper than 10 percent of the wall

thickness shall be accepted for fuel element manufacture. The inactive dead ends shall be wrought stock of AISI Type #316L or 318 stainless steel conforming to the cladding composition, and subject to ASTM Specifications #A-269 or #A-167 wherever applicable.

The dimensional requirements of the fuel tube are illustrated in detail in Figure 1.4. The fuel loading shall be 68.5 grams of UO_2 for the outer fuel tube and 51.5 grams for the inner fuel tube. The outside and inside surfaces of the finished fuel tube shall have a minimum surface finish of 63 rms. micro-inches.

The following quality control tests shall be required on these fuel tubes:

- I. Non-destructive testing (these tests shall be performed on 100% of the fuel elements.)
 - A. All of the fuel elements shall be X-rayed to determine active fuel lengths of the finished tubes, segregation of fuel, fracture and stringering of the fuel particles, and general distribution of the uranium oxide in the matrix. X-ray densitometry analysis shall be made to verify localized fuel distribution. The radiographic techniques shall be detailed in writing by the vendor for the approval of the purchaser.
 - B. Porosity tests conducted as follows:
 1. Remove surface contamination.
 2. Soak in 50% nitric acid solution at room temperature for 24 hours.

3. Evaporate solution and check residue for alpha activity.
 4. Any activity above a standardized background shall be cause for rejection.
- C. Two ultrasonic tests shall be used to evaluate bonding and defects in the fuel tube.
1. An immersed transmission type using Li_2SO_4 transducers operated at ten megacycles. One will be used as the sender and another as the receiver covering a double wall test of the fuel tube. All defects that are greater than that observed for a standard 3/64 inch simulated defect shall be rejected. All tests showing no observable defect of any size shall be accepted. Tubes neither accepted nor rejected shall be tested as follows (2):
 2. An immersed pulse-echo reflection test for single fuel tube wall transmission. One transducer of Li_2SO_4 shall be used as both transmitter and receiver operated at five megacycles. The fuel tubes shall be scanned twice, one in each longitudinal direction. The observable defects of the questionable elements shall be reported to the purchaser for acceptance or rejection.
- D. All tubes shall be dimensionally inspected.
2. The following destructive tests shall be performed on sample quantities of fuel tubes.
- A. Thermal shock to evaluate bond strength.
 - B. Autoclave tests followed by ultrasonic testing to evaluate bonding.

- C. Peel test for bond evaluation.
- D. Metallographic analysis shall be used to establish quality of other tests, core and clad thickness, and fuel distribution.
- E. Chemical analysis to determine fuel composition, fuel distribution, and cladding and diluent composition.

1.1.3 Boiler Core Control Rod

The boiler control rod as shown in Figure 1.11 is cruciform in shape and the active length is 72 inches and is made of 2% boron stainless steel. The blades have a tip to tip dimension of $10\frac{1}{2}$ inches and are $\frac{1}{4}$ inch thick. At the top of the poison section there is a type 304L stainless steel extension and at the bottom of the poison section there is a $\frac{1}{2}$ inch thick 304L stainless steel nose.

All longitudinal welds are made in a fixture in order to minimize twist and warp so that blades of the poison section fit within a 0.293 inch envelope, and those of the extension fit within a $\frac{5}{16}$ inch envelope.

1.1.4 Superheater Control Rods

Superheater control is accomplished by four control rods and drive units. Each of the rods (Figure 1.12) is made up of thirteen poison fingers suspended from a cruciform shaped rack. Each of the poison fingers consists of a series of 2% boron stainless steel cylinders with a $\frac{3}{4}$ inch outside diameter and $\frac{1}{4}$ inch inside diameter brazed to a $\frac{1}{4}$ inch stainless steel tube for a six foot length. Brazing these cylinders to the tube affords a means of cooling the inside of the poison cylinders and adds support to the boron

stainless steel poison section. Each seven foot .030 inch wall stainless steel extension tube is welded to the top of the poison section and is connected to the rack with a quick release pin. Each control assembly can be inserted and removed as a single unit. The rack can also be removed and each of the 13 fingers handled separately. It is not intended that these rods should be scrambled. Rather they will be "run" in under power by the drive unit.

1.2 Reactor Physics

The Pathfinder reactor is a boiling water reactor with an integral nuclear superheater. The boiler region consists of slightly enriched uranium dioxide fuel clad with Zircaloy-2, utilizing water as moderator and coolant. The superheater region consists of highly enriched uranium dioxide - stainless steel cermet clad with stainless steel, also utilizing water as moderator and steam as coolant. The superheater is located in the center of the reactor core with the boiler region surrounding it.

The nuclear properties of the boiler region are typical of lattices which have been tested in critical facilities and operating reactors. The superheater fuel elements differ significantly from those in the boiler because of high metal temperatures and steam coolant. The superheat lattice parameters (U-235 concentration, water to fuel ratio, and cell size) were selected to achieve the following nuclear characteristics:

1. A constant ratio of power between the superheater and boiler during the fuel core life.

2. Low nuclear hot spot factors.

3. Minimum reactivity change associated with flooding the steam passages with water.

The lattice constants of the superheater and the nuclear properties of the combined superheater-boiler core are being experimentally determined in the A-C critical facility.

1.2.1 Nuclear Constants and Neutron Balance

The fast constants were obtained with the MUFT IV program which calculates the Fourier transform of the slowing down distribution. The fast fission factor and the resonance escape probability were determined from analyses of experimental data of similar water-moderated, low enrichment oxide lattices. Thermal cross sections are averaged over a Wigner-Wilkins spectrum using the Sofocate code. Core material volume fractions and nuclear constants are given in Table 1.1. The neutron balance for the operating core at the beginning of the exposure cycle is given in Table 1.2.

TABLE 1.1 CORE PROPERTIES AND NUCLEAR CONSTANTS

	BOILER								SUPERHEATER	
	Lower Half				Upper Half					
Temperature	cold	hot	hot	hot	cold	hot	hot	hot	cold	hot
% Void	0	0	15	30	0	0	20	40	0	0
Σ_f^{Th} (cm ⁻¹)	.0412	.0315	.0310	.0304	.0336	.0256	.0251	.0244	.0429	.0346
Σ_f^{Th} (cm ⁻¹)	.0711	.0533	.0517	.0501	.0614	.0453	.0434	.0411	.0978	.0783
L ₂	3.88	7.35	8.49	9.96	4.04	7.96	9.84	12.6	3.49	6.17
ϵ	1.036	1.042	1.045	1.047	1.028	1.033	1.038	1.043	1.0	1.0
ρ^{28}	.842	.809	.787	.760	.884	.858	.834	.800	1.0	1.0
τ	41.7	57.4	68.8	84.1	38.7	54.7	73.4	101.	95.6	138
σ_a (H)	.252	.197	.194	.191	.261	.202	.199	.195	.235	.186
σ_a (U-235)	508	380	375	368	527	390	385	377	469	357
σ_a (U-233)	2.07	1.62	1.60	1.57	2.14	1.66	1.64	1.61	1.93	1.53
σ_a (Zr)	.171	.134	.132	.130	.177	.137	.135	.133	.160	.126
σ_a (304-SS)	2.13	1.66	1.64	1.61	2.20	1.70	1.68	1.65	1.99	1.58
VF* (H ₂ O)	.5581	.4431	.3368	.3306	.6291	.4994	.4131	.3295	.4146	.3292
VF* (H ₂ O ₂)	.2515	.2515	.2515	.2515	.1987	.1987	.1987	.1987	.00614	.00614
VF* (U-235)	.00222	.00222	.00222	.00222	.00176	.00176	.00176	.00176	.00278	.00278
VF* (Zr)	.1904	.1904	.1904	.1904	.1722	.1722	.1722	.1722	-----	-----
VF* (304-SS)	----	----	----	----	----	----	----	----	.1398	.1398

* VF denotes volume fraction; in the case of water, equivalent volume at 1.0 g/cc.

(1.41)

TABLE 1.2 NEUTRON BALANCE

	<u>Boiler</u>	<u>Superheater</u>
Fast neutrons produced by thermal fissions	.800	.200
Total fast neutrons produced (including fast fissions)	.964	.233
Neutrons lost by fast leakage	.060	.008
Neutrons captured in U-238 resonances	.188	0
Neutrons lost by thermal leakage	.006	.002
Neutrons absorbed in:		
Water	.057	.009
Zr	.013	0
304-SS	0	.037
Xe and Sm	.024	.007
Control rods and burnable poison	.085	.048
U-238 (thermal)	.100	0
U-235	.431	.122

1.2.2 Experimental Results

An extensive program is being carried out in the Allis-Chalmers Critical Facility for the measurement of the nuclear characteristics of the Pathfinder reactor. The reactor used for these experiments is a full scale nuclear mock-up of the boiler-superheater Pathfinder core. In addition, extension of the mock-up and a range of enrichments, water to fuel ratios, core sizes, void contents, and control materials, around the reference Pathfinder core, is provided.

The program of critical experiments consists of three phases: Phase I provides for the measurements of the characteristics of the boiler core alone, in a simple geometry; Phase II provides for the measurement of the characteristics of a superheater-boiler combination core, in a simple geometry; Phase III provides for the extensive measurements of the characteristics of the full scale boiler-superheater core loading in the Pathfinder geometry. Presently, Phases I and II have been completed. The results that have been analyzed to date are summarized in the following tables.

Table 1.3a

SUMMARY OF RESULTS ON BOILER SLAB EXPERIMENTS

Enrichment (% U-235)	W/U	Loading (kg. U-235)	Slab Size (inches)	Void Content	K effective	
					Experimental	Calculated
1.8	3.8	25.2	20.4x27.7x72	0	1.0025	0.996
1.8	5.7	26.4	21.7x34.9x72	0	1.0025	1.0081
1.8	4.8 (voided)	33.8	21.7x44.1x72	15.2%	1.0019	1.0053
1.8	3.1 (voided)	33.3	21.7x35.0x72	18.5%	1.0025	0.9994
1.59	3.8	37.7	21.7x44.1x72	0	1.0010	0.9973

Table 1.3b

TEMPERATURE EFFECTS (1.8% ENRICHMENT)

W/U	Slab Size (inches)	Temperature Coefficient $\% \Delta k/k \times 10^5$		Reactivity Swing from 68 F to 180 F $\% \Delta k/k$	
		at 70 F	at 160 F	Experimental	Calculated
3.8	20.4x27.7x72	-0.6 $\pm$ 0.2	-4.4 $\pm$ 0.2	-0.33	-0.47
5.7	21.7x34.9x72	0.0 $\pm$ 0.2	-0.6 $\pm$ 0.2	-0.1	-0.27

1.2.3 Neutron Spectrum

The thermal neutron spectrum is important in reactor analysis because of its effect on the average thermal cross sections and on reactivity coefficients. It assumes additional importance in the nuclear design of Pathfinder since spectral effects contribute to the superheater flooding coefficient and to the flux shape at the superheater-boiler interface. Listed in Table 1.4 are some results from Wigner-Wilkins flux spectrum calculations. Neutron energies corresponding to the peak of the thermal flux spectrums are given for various core regions. It should be noted that these peaks are nearly the same for each core region. This does not mean that the spectrums are identical but does show that sufficient thermalization takes place throughout the core; including the steam-cooled, water moderated, superheater.

TABLE 1.4

Energies of maximum thermal flux, $T = 489^{\circ}\text{F}$

Superheater Region:

Operating-----	0.050 ev
Steam Passages Flooded-----	0.048 ev

Boiler Regions:

Lower Half-----	0.049 ev
Upper Half-----	0.048 ev

The neutron flux, as a function of energy, is shown graphically in Figure 1.13. Curve A, typical of the boiler region, represents the spectrum in the 64 rod section with 20% steam voids. Curves B and C represent the superheater spectrums with flooded and with voided steam passages respectively. This spectral change, resulting from flooding to the superheater steam passages with saturated water, corresponds to a k_{00} change of about 0.1%.

1.2.4 Flux and Power Distributions

The flux and power distributions in a uniformly loaded core, where the moderator is permitted to boil, tend to be asymmetric because of the distribution of voids. This normally flattens the radial power distribution (no inlet orificing) and increases peaking of the axial power distribution. The latter usually becomes worse because of control effects. In the absence of power or flux shaping devices, the hot spot factors can become high in a uniformly loaded core, limiting the total power generation. In particular, reduction of the overall hot spot factor is obtained by varying the axial fuel loading. In addition, the centrally located superheater decreases the radial power distribution because of its relatively low reactivity.

10. The power distribution for Particulars are shown in Figure 1.13. These power distributions are based on the core design of CANDU-600, and the distribution shown in Fig. 1.13. The power distribution is based on the core design of CANDU-600, and the distribution shown in Fig. 1.13. The power distribution is based on the core design of CANDU-600, and the distribution shown in Fig. 1.13.

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The axial maximum-to-average power ratio for the boiler region, as obtained from this plot, is 1.44.

The average radial power distribution is shown in Figure 1.15. It is seen here that the low reactivity superheater ($k_{\infty}=1.1$) located within the relatively high reactivity boiler region ($k_{\infty}=1.25$) causes significant shaping of the radial power distribution. The maximum-to-average power ratio of this distribution is 1.19. Discontinuities in the superheater region are due to non-uniform lattice spacing.

Other factors which affect the power distribution are local flux peaking at water channels, peaking at the superheater-boiler interface, and control rod effects. The core is being designed so as to minimize the size of water channels within the core. The largest expected channel occurs in the control rod guide tube when the rod is withdrawn. This produces an effective water channel of 1/2 inch and a peaking factor of about 1.2 in the fuel. At the superheater-boiler interface, peaking effects are expected to be small since there are essentially no water channels present and the nuclear properties of the two regions are matched. Flux distortions due to necessary control rod insertion will be minimized by rod programming. Table 1.5 lists magnitudes of factors which contribute to the hot spot factors of both the boiler and the superheater regions.

TABLE 1.5

Core Hot Spot Factors

	<u>Boiler</u>	<u>Superheater</u>
Axial, Radial, Control Channel	2.5	2.0
Flux Uncertainties	1.1	1.1
Fuel Distribution	1.1	1.15
Overpower	<u>1.1</u>	<u>1.1</u>
	3.3	2.8

When the reactor is cold and clean, the power distribution is heavily peaked in the boiler with respect to the central superheater. This is due to the higher reactivity of the boiler and its relatively large water to fuel ratio at cold clean conditions. With the core at operating temperature, but no voids in the boiler, the power distribution flattens somewhat, radially, especially in the lower core regions. The flattening is caused by the large loss of reactivity in the boiler regions. The reactivity in the superheater does not vary appreciably with increased temperatures. Under power operating conditions, the power distribution is that shown on Figure 1.14. Because the axial fuel loading in the boiler is such that a reasonably constant water to fuel ratio is obtained along the core height, this radial power distribution is maintained at the various axial levels.

Because of the importance of local and gross power distributions in power reactor operation, considerable effort will be expended during the full scale critical experiments to verify and, if possible, to reduce these factors.

1.2.5 Reactivity Coefficients

The reactivity of the core is affected by operational power changes; namely, changes in temperature, moderator voids, and fission product concentration. The temperature dependence may be divided into a moderator and a fuel temperature effect. The moderator temperature and void coefficients are related closely to the choice of lattice since the ratio of water to fuel is an important factor in determining their magnitude and sign. The boiler region lattice has been designed to be in the "dry" range (H_2O to U-235 ratio of about 100) to achieve high specific power and economical operation, thereby producing negative coefficients. The fuel temperature coefficient as caused by the Doppler effect, or broadening of the resonance peaks of U-238, is also dependent on the lattice. However, it is a negative effect.

The reactivity change of the system caused by temperature, voids, and xenon and samarium poison are summarized as follows:

	<u>Reactivity Change</u>
Temperature (moderator)	1.8% $\Delta k/k$
Voids (boiler core)	2.3 $\Delta k/k$
Doppler effect (boiler core)	1.3 $\Delta k/k$
Xenon and samarium equilibrium poisoning	<u>3.5 $\Delta k/k$</u>
Total change	0.9% $\Delta k/k$

In the superheater region, where boiling of the moderator is suppressed, moderator temperature and xenon poisoning are the primary causes of reactivity variations during power changes. The infinite multiplication constant of the

cold, clean superheater, with steam passages voided of water, is 1.05. Increasing the moderator temperature to its operating level changes k_{∞} to 1.04. An increase to equilibrium xenon concentration reduces the k_{∞} of the superheater to 1.01. The fuel temperature in the superheater has little effect on the reactivity. A negligible fuel temperature coefficient of $-1 \times 10^{-5} \% \Delta k/k_F$, due to density changes and elongation of the superheater fuel tubes, has been estimated.

A reactivity change which is peculiar to the superheater cores is due to the removal or addition of water within the steam passages. The steam passages are here defined as the spaces between: (1) the central poison rod and the first fuel tube, (2) the first and second fuel tube, and (3) the second fuel tube and the inner wall of the insulating tube. These passages will normally be flooded with water during shutdown and filled with steam during operation. There is, therefore, a significant reduction of the moderator in the superheater between the shutdown state and the operating state. Conversely, an accidental flooding of these tubes at any time will significantly increase the moderator content of the superheater.

The reactivity change due to this will be referred to as the "flooding coefficient". Both theoretical analyses and experimental work (using a small superheater fuel sample in the Penn State Research Reactor) have shown that this flooding coefficient will be small. The change in k_{eff} due to superheater flooding is approximately $\pm 0.2 \% \Delta k/k$. This

reactivity change results from the decrease in neutron leakage, the decrease of thermal utilization, and the softening of the thermal spectrum. With burnup the value of this coefficient will decrease and become slightly negative.

The foregoing discussion dealt with reactivity changes resulting from variations in the amount of water present in the steam passages of the superheater fuel elements. Reactivity changes are also caused by variations of water present in the annular insulation spaces of the superheater fuel elements. These changes are minor and do not present a safety problem. The reactivity effect caused by the forcing of water out of these insulation spaces, occurs slowly since it is dependent on a slowly increasing fuel temperature. The filling of these spaces with water, occurs more rapidly since flooding of the superheater causes a cooling of the insulation spaces, thereby permitting water to be drawn into them.

Operational procedures provide for safe draining of the superheater with the reactor shut down. Accidental draining of the superheater is analyzed in Section 7.

If the superheater were to be flooded during operation, the reduction of temperature in the insulating space will cause water to be drawn up into this space. It should be noted that the flooding of the superheater during operation is an unlikely event. Large volumes of steam pass through this section during operation making the residence of water, for any appreciable time, very improbable.

The reactivity changes encountered in the superheater operation, particularly the flooding coefficient, will be given considerable experimental attention during operation of the Allis-Chalmers Critical Facility. It is not anticipated that this will be an operational problem for the Pathfinder Plant. If the measurements indicate larger changes than are now expected, moderate redesign of the superheater lattice can be made to alter the magnitude of this coefficient.

Based on reported experimental data¹ the Doppler effect for the highly enriched superheater is expected to be very small. This is because the superheater spectrum is thermal, its flux peak energy being about 0.050 ev., and the U-235 loading is quite dilute, making the self shielding factors small, thereby reducing the strength of the nuclear Doppler effect.

1.2.5 Reactivity Requirements and Control

The reactivity demands made on the control system may be divided into the short term effects, such as previously described and the long term burnup effects. The initial phase of operation, that of increasing reactor power to the design level, will result in a reactivity decrease of $8.9\% \Delta k/k$, at xenon and samarium equilibrium. An additional $3\% \Delta k/k$ allowance is provided for shut down safety margin. The reactivity allowance for burnout in fuel and building up of non-saturating fission product poisons is $5\% \Delta k/k$.

¹ Nuclear Science & Engineering, Vol. 3, Page 107

Additional burnup control is provided through the use of a burnable poison. Since the superheater is a burner rather than a converter, it is desirable to locate most of the burnable poison in this region to aid in optimizing its reactivity characteristics throughout the core life. The worth of the boron poison in the reactor is about 7% $\Delta k/k$. Although the reactivity of the superheater region may increase for a period, early in the superheater life, the reactivity in the boiler region continually decreases and at no time will the superheater or the combination boiler-superheater core contain sufficient reactivity to reduce the 3% $\Delta k/k$ shutdown margin previously specified.

1.3 Pressure Vessel

The reactor vessel and support structure is designed in accordance with the ASME Boiler and Pressure Vessel Code for Unfired Pressure Vessels, Section VIII and Section IX, 1956 Edition; the American Standard Code for Pressure Piping, 1955 Edition, and good design engineering practices.

The design basis was by mutual agreement between Allis-Chalmers, the Northern States Power Company and the State of South Dakota.

The pressure induced stresses in the reactor vessel are limited by design in accordance with the ASME Unfired Pressure Vessel Code, Section VIII. These values are used with the specified weld efficiencies to determine the wall thickness. Special attention was given to any nozzle, closure, or penetration not covered in detail by the code. A maximum allowable discontinuity stress index of 1.5 was used. (The stress index is the ratio of the peak stress due to the discontinuity to the average circumferential membrane stress in the cylindrical shell.) This value is considered to be both conservative and realistic, and was also used as the design limit in evaluating the reinforcements around all penetrations.

Thermal stresses due to heat losses are minimized by adequately covering the vessel with thermal insulation. The expected temperature drop across the vessel wall is small, resulting in a thermal stress of less than 100 psi. Radiation induced thermal stresses are minimized by the large water passages surrounding the boiler core. The maximum radiation-induced thermal stress occurring in the vessel adjacent to

the core centerline is calculated to be 650 psi.

Radiation levels determined by physics calculations for the reactor pressure vessel were employed in a thermal stress and radiation damage analyses. It was found that the vessel materials will not be subject to any significant damage or property changes during the life of the reactor.

The vessel (Figure 1.16) has an overall diameter of 11 feet 6 inches and an overall height of 30 feet 6 inches. The vessel is mounted on support columns within the shield cavity and will be covered with 3 inches of thermal insulation.

Prior to the start of vessel fabrication a complete design report was submitted to the Hartford Steam Boiler Inspection and Insurance Co. for review. The Hartford Company is the ASME Code Inspection agency to the Allis-Chalmers Mfg. Company, the fabricator of the reactor vessel. On January 11, 1960 approval of the design was made by the Hartford Steam Boiler Inspection and Insurance Company.

The vessel is fabricated from 3 inch thick composite plate, consisting of a SA-212 grade B carbon steel strength member clad on the inside with type 304L stainless steel 1/4 inch thick to provide adequate corrosion protection. The cladding is integrally bonded to the carbon steel backing material by a nickel interface layer. The bond is obtained by hot rolling the composite plate. A bonding strength of 45,000 psi is claimed by the manufacturer of the clad plate, however, no credit is taken in the design of the pressure vessel for any mechanical strength of the cladding. Under operating conditions the corrosion rate is expected to be 0.0001 inches per year. The 1/4 inch

cladding thickness is sufficient to take care of localized corrosion rates much higher than average for the life of the vessel.

The vessel will be cylindrical in geometry with a hemispherically dished head at the top and a hemispherical head at the bottom. The lower head includes a 4 foot inside diameter cylindrical water inlet section. The water inlet section contains three nozzles for 22 inch water recirculation inlet piping as well as a nozzle for a 16 inch steam pipe. The lower head proper contains three nozzles for 22 inch water recirculation outlet piping and a nozzle for an 8 inch feedwater line.

The recirculation water inlet nozzles are fabricated from SA-212 Gr. B plate and then weld-clad to a thickness of 1/8 inch on the inside diameter with type 304L stainless steel. The recirculation water outlet nozzles are forged of ASTM A-105 Gr 11 carbon steel and weld-clad on the inner surface with type 304L stainless steel. The main steam outlet nozzle is designed as a thermal sleeve because the steam outlet temperature is 825 F compared with the 489 F water within the vessel. The main nozzle forging is ASTM A-105 Gr 11 carbon steel weld-clad on the inside surface with type 304L stainless steel and the thermal sleeve is ASTM A-335 Gr P-22 Chrome-Moly steel weld-clad on the inside surface with type 304L stainless steel. The feedwater inlet nozzle is also of the thermal sleeve design since the inlet water temperature is 356 F. The main feedwater nozzle forging is of ASTM A-105 Gr 11 carbon steel weld clad with type 304L stainless steel and the thermal sleeve is type 304L stainless steel. All thermal sleeve materials were selected to match the joining piping.

The upper head contains the removable cover (Figure 1.10) with a clearance diameter of 7 feet 7 inches. This provides sufficient clearance for removal of the core grid plate, which is the largest full diameter component removable from the vessel. The removable cover contains 20 control rod mounting tubes.

The vessel cover is located accurately with respect to the vessel by means of three positioning studs in the vessel flange. This feature is necessary to align the control rod drives and control rods with respect to the channel provided for them in the reactor core. The mounting tubes are fabricated of Inconel pipe to match the thermal expansion of the vessel cover and still provide corrosion protection.

The cover will be fastened to the pressure vessel by bolts. Sealing is accomplished by two corrugated, type 316 stainless steel jacketed, soft iron core gaskets. The inner gasket is constructed of a .019 inches thick type 316 stainless steel shell and a 1/12 inch soft iron corrugated core. Gasket size is 91 1/2 inches I.D. x 94 inches O.D.

The outer gasket is constructed of .019 inches thick type 316 stainless steel shell and .019 inches thick soft iron corrugated core. Gasket size is 94 1/2 inches I.D. x 96 1/2 inches O.D. The inner gasket carries about 75% of the bolt load and the outer 25%.

To produce the uniformity, hydraulic bolt tensioners will be used to remove or replace the bolts. Two bolts, 90° apart on the bolt circle, will be tensioned at a time to insure uniform compression

of the gaskets and therefore leak-tightness. The space between the gaskets is provided with a leak-off, the discharge from which is monitored and will indicate a leak if one should develop. In addition, the second gasket provides a backup in case the first one should fail.

The main closure bolting consists of 48 3-inch diameter studs designed in accordance with standard practice. The normal stress in the bolts under operating conditions of temperature and pressure is 27,000 psi by calculation.

Above the vessel is a water shield pool, approximately 18 feet deep. The vessel cover serves as part of the bottom of the shield pool. The cover is separated from the shield pool water by a layer of metallic thermal insulation in a stainless steel cover bolted to the cover. A removable seal prevents shield pool water from penetrating to the vessel at the bolting circle. This seal consists of doughnut shaped canned insulation provided with two rubber "O"-ring gaskets (Figure 1.17). The seal ring spans the bolting circle with the inner "O"-ring resting on the cover and the outer "O"-ring on the vessel insulation shroud. A leak-off connection drains the water from the space beneath the seal. The space between the vessel and shielding concrete surrounding it is sealed off by a stainless steel expansion bellows. The bellows is designed for a one inch differential expansion between the vessel and shield pool floor.

Two nozzles for liquid level indicator pipes are provided in the cylindrical portion of the pressure vessel. On the inner wall there are welded mounting lugs for supporting the steam separators. At the base

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In order to insure that the vessel is of sound construction the following inspection techniques were employed. Allis-Chalmers was responsible for the preparation of inspection procedures and the supervision of all required inspections.

1. Internal surfaces. To detect any possible cracks in the stainless steel cladding on the carbon steel backing material of the pressure vessel, a dye penetrant check of the inner surface of the vessel was performed. A complete check of the surface prevented possible corrosive attack of the carbon steel backing material through minor cracks in the stainless steel. All cracks and defects due to manufacturing were mechanically removed and repair welded.
2. Clad plates. All stainless steel clad plates were inspected by ultrasonic techniques at the mill and spot checked upon receipt by Allis-Chalmers. All plates were found to be sound and free of defects before the start of fabrication.
3. Forgings. All forgings were inspected by using ultrasonic techniques to insure a sound forging free of any defect which could cause failure of the part, or impair a finished gasket surface or machined fitting surface.
4. Castings. All castings were inspected in accordance with the specifications of ASTM E-71 "Tentative Industrial Radiographic Standards for Steel Castings." All castings met as a minimum either Class 1 or Class 2 standards depending on the wall thickness.

5. Welds. The first and last passes of any weld were not peened. All welds were 100% radiographically acceptable according to the standards of UW-51 of Section VIII of the Pressure Vessel Code. The first two passes and the final pass were magnafluxed, checked with dye penetrant or equivalent.

Welds which showed defects as evidenced by the various tests and welds which were unacceptable as evidenced by radiographic examination, were repaired by mechanically removing the defects and rewelding. No arc or flame cutting was permitted. This repair procedure was adhered to on both welds and base metals.

6. Hydrostatic test. A hydrostatic test was conducted in accordance with the requirements of paragraph UG-99 of the Unfired Pressure Vessel Code Section VIII at a pressure of 1050 psig which is $1 \frac{1}{2}$ times the maximum allowable working pressure. The vessel was tested at 70 F which was at least 50 F above the nil-ductility temperature as determined by the Charpy-A 15 ft.-lb impact test.

Allis-Chalmers supervised the inspection of pressure vessel materials and components as follows:

- a) Materials were purchased to rigid Allis-Chalmers specifications outlining physical and chemical requirements. Test bars were taken from the materials and certified reports by suppliers were submitted to Allis-

Chalmers attesting to material properties. Clad plate materials were examined by ultrasonic techniques for voids between clad and backing material upon receipt at Allis-Chalmers.

- b) Components were also purchased to Allis-Chalmers specifications, and inspections of such components were handled by regional inspectors during fabrication phases. Items such as clad and formed heads received a complete ultrasonic inspection prior to leaving the supplier's plant.
- c) The type of inspection and the point in the fabrication sequence at which the inspection was to be conducted, is called out in the "Manufacturing Procedure for Reactor Pressure Vessel Components Made From Stainless Steel Clad Plate" Specification (43-101-129) and others.

1.4 Control Rod Drives

All control rod drive mechanisms are located on the vessel cover and submerged in the shield pool. The drive housing extends 11 feet above the vessel cover. Power is supplied to the drives through water-proof cables from a junction box above the shield pool water level.

The drive mechanism is of the rack and pinion type. The general layout of the driving components is shown in Figure 1.18.

Motive power to the drive is supplied by a 3-phase reversible induction gearmotor with integral spur gear box and mechanical brake. The output from the gear motor is transmitted through a fail-safe

magnetic scram clutch and drive shaft to miter gears which change the driving direction at right angles. The power is then transmitted by a pinion shaft through two seals, one high pressure and one low pressure, to a pinion driving the round control rod drive rack. The control rod drive rack, guided by a back up roller and a Stellite bushing, is latched directly to the extension of the control rod.

The scram clutch disconnects the drive shaft from the gear motor and allows the control rod, with rod extension and rack attached, to drop into the reactor for an emergency shutdown. A scram spring is used to assist gravity thus enabling the approximately 160 pound rod to drop the full stroke of 72 inches in approximately one second. Signal delays are small in relation to rod drop time, thus the total scram time is approximately one second. A hydraulic dash pot surrounding the rack stops the mechanism and the control rod during the last 6 inches of the stroke.

To eliminate a circulating steam path through the control rod drive which would result in excessive heat losses to the shield pool, a conventional piston type accumulator is mounted on the control rod drive (refer to Fig. 1.18). A steam line from the pressure vessel is connected to the bottom of the accumulator and a water line connects from the upper end of the accumulator to the top of the drive rack housing. The piston of the accumulator forms a seal between the upper and lower connection points. On scram, when the rack is withdrawn from the rack housing, the pressure in the housing and in the upper end of the accumulator is reduced. Reactor steam on the lower

side of the accumulator piston then pushes the piston up and forces the water from the accumulator into the rack housing.

After scram a spring forces the piston downward thus reducing the pressure in the rack housing and allowing steam to flow through the rack bushing from the reactor. The steam condenses in the housing and allows the accumulator piston to return to its normal position.

A servo position indicating transmitter is driven off the main drive shaft by a gear train, indicating the position of the control rod at all times. Limit switches also driven by the gear train prevent over-travel of the rod in either direction and provides, by means of lights on the control panel, an independent way of sensing control rod position.

The housing enclosing the gearmotor, magnetic scram clutch, position indicator assembly, drive shaft bearings, the conventional bearing for the pinion shaft, and the miter gears, is filled with a transformer grade oil. The very good electrical insulating qualities, low viscosity, and radiation stability of this oil provides a favorable and non-corrosive environment and permits the use of standard materials. The low pressure rotary face seal on the pinion shaft prevents leakage of the transformer oil from the drive housing. Any leakage past the seal goes into a monitored seal drain.

The basic design features of the driven components are shown in Figure 1.18. Reactor steam is permitted to condense in the rack housing which is then filled with water at about 150F and 600 psig. The Stellite bushing which surrounds the round rack acts as a seal

to prevent water leakage back into the reactor vessel. The high pressure rotary face seal on the main drive pinion shaft prevents leakage of the high-pressure water from the rack housing. Any leakage past the high pressure seal goes to the monitored seal drain, the same seal drain that carries away leakage from the low pressure rotary face seal on the drive housing.

All connections to the control rod drives are made above the level of the shield pool. The connection lines are brought from the side of the shield pool to the drives by a tilt-up bridge. All connections are removed from the drives for refueling operations and the tilt-up bridge is moved out of the area required for the traveling bridge cranes used to handle fuel elements (refer to Fig. 1.19).

A scram assist spring is located in the upper end of the slotted inner containment tube. The scram assist spring is twenty inches long and is compressed to a 10 inch length as the drive rack moves to its upper position. The stored energy of the scram spring is used to assist gravity in driving the control rod into the core during scram.

The control rod is attached to the rack by means of a bayonet latch. The bayonet latch is locked or unlocked by rotating the round rack 45 degrees. Rotation of the rack can be achieved only by the use of a special unlatching tool which turns the inner containment tube, and, therefore, the rack. While the reactor is operating, the inner containment tube is locked in place, thus preventing rotation of the rack.

The gearmotor is controlled selectively by either a manual pushbutton or an automatic servo control. The low inertia rotor of the motor stops very rapidly because of an integral mechanical brake. Coast is less than 1/8 inch.

Because of the speed limiting characteristics of the A-C motor, a dangerous withdrawal rate for the control rods can never be reached. The driven velocity of the control rods is 75 inches per minute.

Whenever it is required to remove the vessel cover the control rods are disconnected and remain in the core. The procedure is as follows:

1. A special tool is used from the drive bridge to engage the square top of the rack turning mechanism. By pushing down the ball lock is released and the rack may be turned 45° to disengage the control rod from the latch.
2. The rack is then raised a few inches to disengage it completely from the rods.
3. The rack is then rotated back to its original position and driven up until its bottom end is above the plane of the lowest edge of the pressure vessel cover. The rack is then rotated 45° to lock it in position so that the power leads can be disconnected without dropping the rack.
4. The cover is removed and placed in the storage space provided for it in the shield pool. Support bosses are provided on the cover in order to protect its seal surfaces during storage in the pool.

The control rod drive housing, including the rack and drive is removable as a unit by unbolting it from the vessel cover. This operation can be performed without removing the vessel cover providing that the reactor is shut down and depressurized and the vessel is flooded.

In the design of the control rod drives the seal requirements have been evaluated to utilize the developments in the art to the fullest extent. The design contains two rotating seals and one reciprocating buffer bushing. The first rotating seal isolates the reactor pressure from the seal drain and operates in water at about 110F, the second rotating seal separates the drive motor housing from the seal drain and operates in transformer oil at about 120F, max. The buffer bushing to handle reciprocating movement is located at the base of the rod drive and limits leakage in the event of a seal failure in the control rod drive mechanism. The normal leakage past the main seal should not exceed 0.10 pounds of water per hour.

If the control rod drive shaft seal should rupture, the amount of leakage from the ruptured seal would be limited by the buffer bushing. It is anticipated that with full reactor pressure across this buffer bushing, the leakage would not exceed 160 lbs/hr of saturated steam. The leakage would be removed from the seal area by a drain line to a low pressure point. The calculations of maximum leakage rate assumed no back pressure on the buffer bushing and is, therefore, the "worst case" answer.

Excessive leakage from a seal would necessitate reactor shutdown to replace the faulty seal.

Provisions in the control rod drive design to handle various unusual conditions are noted below:

(1) Control rod stuck in an intermediate position

Step 1 - Energize drive motor in the down direction and attempt to drive the rod down.

Step 2 - If the rod cannot be driven down then the reactor must be shut down to atmospheric pressure and the pressure cover unbolted from the top of the drive. With the drive cover removed a long handled socket wrench is employed to remove the rack retainer. The control rod drive can now be removed from the vessel cover or the entire cover assembly may be removed provided all the rest of the control rods have been unlatched. The latch coupled to the stuck control rod would remain in the reactor and would have to be worked on from directly above the shield pool with long handled tools.

Step 3 - As an alternate to Step 2 the rack retainer may be tapped carefully from above to drive the rod in.

(2) Jammed gearmotor or bearing above the magnetic clutch with rod in any intermediate position.

Step 1 - Shut down the reactor and press the scram switch to de-energize the magnetic clutches.

Step 2 - Unlatch the rod in the normal manner.

(3) Jammed bearing or gear below the magnetic clutch with rod
in any intermediate position.

Step 1 - Energize drive motor in the down direction and
attempt to drive the rod down.

Step 2 - Remove the drive cover and apply force to the top
of the rack retainer until the drive rack is forced
to the full down position. Sufficient force may
be applied without damage to the adjacent fuel
elements. After the rod has bottomed the rack
would have to be forced to rotate to the unlatched
position and the damaged drive would be removed
before the vessel head since the rack would be
fully extended.

NOTE: It may be necessary to lift the entire drive
slightly to determine if the control rod is
jammed as described in (1) above.

(4) Latch frozen to the control rod for any reason in the "latch"
position.

Step 1 - Drive to maximum overtravel position and try to
turn the rack.

Step 2 - If necessary approximately 1000 ft. lbs. of torque
may be applied to the latching tool to shear the
shear pin provided in the latch coupling.

Step 3 - Remove the cover in the normal fashion and remove
the control rod and latch as a unit.

The four superheater control rod drives differ from the boiler rod drives as follows:

1. The superheater rod drives do not contain a scram spring since they are not normally scrammed. Because of the multiple-coupled rod design these rods are driven in at the normal 75 inches/minute.
2. A separate manual clutch release switch is provided on the control panel to be used only in the case that the drive motor or gearbox jams.
3. Upon loss of power the clutch releases and the superheater rods fall free into the reactor.
4. The superheater design dictated a design in which the latch lugs are at 45° to the latch lugs in the boiler region drives.

1.5 Reactor Internals

1.5.1 Coolant Circulation (Refer to Figure 1.16)

The feedwater is pumped into the reactor vessel at 365 F and is then distributed in the reflector annulus through a distribution ring consisting of 6 inch diameter pipe containing 250 - 3/8 inch diameter holes. The holes are located so that the feedwater is directed away from the vessel wall to insure against thermal shock. Mixing with the recirculation water is thorough because of the turbulence in this region and the resulting exit water temperature is 485.8 F. In the recirculation loops, pumps raise the water pressure to 642 psia. The coolant then enters the bottom of the reactor vessel and flows upward through the boiler core where its temperature is raised to 488.9 F and part of the water changed to steam. The coolant emerging from the boiler core is a mixture of steam and water. Most of the steam is separated from the water at the surface. The water and some entrained steam flows into steam separators located in the reflector annulus. Here the remaining steam is separated from the water. The water then flows down through the reflector annulus, thus completing the recirculation cycle.

The steam flows upward into the steam dome entering the superheater chimney through steam dryers which removes excess moisture from the steam. The steam flows down through the superheater region where its temperature is raised to 825 F and out of the reactor through a 16 inch steam line. From the reactor vessel the steam flows directly to the turbine.

1.5.2 Boiler Core Layout

The reference boiler core layout is shown in Figure 1.20. This view shows one-half of the boiler region that consists of a total of 96 boiler fuel elements spaced as shown. The boiler fuel boxes are a nominal 5.312 inches square and the average water gaps between elements are as shown. The boiler core contains 16 cruciform shaped control rods nominally 1/4 inch thick with 10-1/2 inch wide blades spaced in the core as shown. The boiler core elements are nested around the superheater baffle in the center, and extend in the radial direction to the boiler baffle at the outside. There is no permanent structure within the limits of the boiler core fuel region.

1.5.3 Superheater Core Layout

The reference superheater core layout is shown in Figure 1.21. This view shows one-half of the superheater region that consists of 468 tubes 1.080 inch outside diameter surrounded completely by a protecting baffle. Of the total number, 415 tubes contain fuel elements, 52 contain control rods and one contains the neutron source as shown in Figure 1.21. The tube spacing within the superheater region varies slightly to control the amount of moderator around the fuel elements, however, the center-to-center spacing is approximately 1-1/4 inches to 1-3/8 inches. The superheater baffle is 1/4 inch thick stainless steel slotted along its length in the tube region to prevent pressure differences and resulting baffle deflections between the boiler and superheater regions.

The superheater baffle serves as a protection for the superheater tubes and as the structural support for the superheater assembly.

1.5.4 Steam Separators

The steam separating device (Refer to Figure 1.22) to be used in the Pathfinder reactor vessel is essentially a 10 inch tube 102 inches long made of 12 ga (.109 in), 304 stainless steel. The inlet nozzle, through which the steam and water mixture enters, opens to a tangential slit in the tube 1.8 inch wide and 54 inches long. The inlet to the slit is inclined at 60° with the vertical.

The separator outlet nozzle is cone shaped and reduces from 10 inches to 4 inches in diameter over a 24 inch length. The conical surface has 6 equally spaced triangular-shaped openings. These openings are 1-1/2 inches wide at the bottom and are 24 inches long. A 4 inch disc closes off the bottom of the separator and forces all recirculation water to go out through the triangular slits.

The top of the steam separator has a 10 inch to 4 inch reducer and a 4 inch diameter section rises into the steam dome area.

Under normal operating conditions the total reactor recirculation flow rate is 60,000 gpm. and 616,125 lbs. per hour of steam is produced. About 20% of the steam produced will be handled by the separators. There are 45 separators, each of which handles 1/45 of the total flow (1330 gpm). Steam carry-under into the recirculation pump suction is about 0.1% by volume. Total maximum pressure drop through the separators is 4.5 feet of water at normal operating conditions.

1.5.5 Steam Dryers

A total of 26 steam dryers are located within the reactor vessel above the steam water interface. Eight dryers, each about 16 inches in diameter, are located directly above the boiler region and are removed as an assembly for refueling. The remaining eighteen are approximately 17-1/2 inches in diameter and are arranged above the downcomer annulus in a permanent structure. The steam dryers contain multiple entrance vanes arranged around the periphery to impart a rapid rotary motion to the steam as it passes through the unit. The developed centrifugal force removes the entrained moisture which is returned to the liquid surface by drain pipes from the dryers. The steam at the entrance to the dryers contains about 1/2% entrained moisture and approximately 99% of this entrained moisture is removed by the dryer.

1.5.6 Boiler Element Hold Down Structure

The boiler element hold down structure is essentially a welded grid assembly of stainless steel bars positioned on the top of the boiler baffle. The bars of the grid bear on the tops of the boiler fuel elements and pre-compress the thermal expansion springs in the elements. The hold down structure also contains the upper control rod guide tubes used to shield the control rod from hydraulic loads due to the coolant flowing toward the downcomer region. Pads mounted at the base of the upper guide tube will engage the tops of adjacent fuel elements and align them to form the proper width control rod channel.

The hold down structure is locked firmly to the top of the boiler baffle during normal operation but is removed during refueling operations. The hold down structure is stored within the shield pool during the refueling operation.

1.5.7 Boiler Baffle

The boiler baffle is fabricated of stainless steel type 304L. It is circular in shape and completely enclosed around the outer periphery. The top and bottom end plates are machined on the inside to fit the core shape. The boiler baffle is 7 feet 5 inches high and 6 feet 7 inches in diameter. Rigidity of the structure is maintained by suitable structural angles and gusset plates. The boiler baffle is fixed directly to the boiler grid plate by bolting, and aligned by dowel pins. The top and bottom end plates are machined flat for mounting purposes. The boiler element hold down structure is locked to the top of the boiler baffle after the core is loaded with fuel. The boiler baffle is a permanent piece of reactor equipment which is removed only in the case of major core change or repair.

1.5.8 Structural Support Members

a) Boiler grid plate:

The grid plate consists of 2 inch thick stainless steel type 304L top and bottom plates welded around the periphery to a 6 inch high spacer ring to form an 8 inch thick sandwich assembly. Nozzles, as fuel element receptacles, are formed in the grid plate by welding individual pipes between the top and bottom plates. The nozzle pipes are then accurately bored to the proper dimensions and location. This type of fabrication is extremely rigid and rugged. The grid plate is bolted and doweled to the grid plate support core which is welded permanently into the pressure vessel.

b) Steam separator support shelf:

The separator support shelf is a stainless steel type 304L plate fabrication mounted on brackets welded directly to the lower portion of the vessel. Differential thermal expansion due to material differences between the pressure vessel wall and the support shelf are handled by mounting the shelf on the vessel brackets with a special eye bolt clamp, thereby allowing the shelf to move independently of the vessel wall. The separators rest on flat counterbores in the shelf surface, thus permitting accurate location without need for close manufacturing tolerances. The steam separators are aligned and held down at the top of a separator group assembly.

c) Steam dryer supports:

The eighteen permanently mounted steam dryers are supported from the vessel wall by brackets similar in design to those employed on the steam separator support shelf. In addition the support plate is held down firmly around the outer periphery since the pressure drop across the steam dryers exerts an upward force on this plate assembly.

The eight dryers directly above the boiler region are mounted in a removable structure supported from lugs on the inside diameter of the main flange opening. Lugs mounted on the top cover of the reactor vessel will lock the assembly in position. Removal of the vessel cover releases the dryer assembly so it can be readily removed from the reactor. During the refueling operation the steam dryer assembly is stored in the shield pool.

1.6 Recirculation System

1.6.1 Recirculation Pumps

The reactor recirculation system consists of three loops, each containing a 20,000 gpm. pump capable of developing a 71 foot head. The purpose of these pumps is to provide forced circulation of coolant and moderator through the boiler core and to provide an auxiliary means of reactor control, since the amount of moderator passing through the core can be varied by throttling recirculation flow with butterfly valves. Flow can also be varied by operating just two of the three pumps.

Cavitation in the recirculation pumps is not expected to be a problem, since ample NPSH is available for all operating conditions. The total suction head available at normal reactor conditions is 54 feet. (44 feet of this total is produced by introducing 356 F feedwater which subcools the reactor water from 488.9 F to 485.5 F, and approximately 10 feet is produced by static head.) Since the pumps require only 24 feet of NPSH for normal operation at design flow, an excess of 30 feet is available. During short operating periods such as startup or shutdown, this value may be lowered to 14 feet. Even when feed-water flow is reduced to its minimum value (100 gpm return from the reactor water purification system) the pumps will have sufficient NPSH by virtue of the subcooling effect of the 100 gpm, plus additional static head secured by raising the reactor water level.

The recirculation pumps are of the mixed-flow type and have 22 inch diameter suction and discharge nozzles. The pumps are mounted on special trunnion support structures which allow freedom of movement in the horizontal

plane. This type of mounting minimizes external forces on the pump casing due to thermal expansion of the piping. The impeller is bolted and keyed to the overhung motor shaft. Supplementary support for the shaft is provided by a fluid piston bearing which extends from the lower end of the impeller.

A floating, bushing-type shaft seal prevents leakage of the primary fluid to the atmosphere. Reactor grade water at 100 F is injected to these seals by one of two positive displacement pumps. Each seal receives 7 gpm under normal operating conditions, 5 gpm flowing into the recirculation pump casing through a labyrinth bushing, and 2 gpm leaking through the seal to a sump. As additional safety devices, a mechanical, spring loaded face seal and a labyrinth bushing are provided just before the shaft emerges from the seal housing. Two gas charged accumulators in the seal water system ensure that the flow of seal water will be maintained in the event that automatic transfer takes place from one seal injection pump to the other.

Each pump is driven by a 400 HP, 2300 volt, 3 phase, 600 rpm induction motor. The motors have thermostats in the windings with normally closed contacts for high temperature overload protection. Each motor bearing housing is equipped with a thermocouple type temperature indicator. In addition, the bearings are protected by a vibration detecting device that will produce an alarm signal under abnormal frequency or amplitude conditions.

The pump casing is cast ASTM A-351-57T grade CF-8 stainless steel. The casing is designed to permit removal of the pump motor and rotating parts as a unit when the casing cover is unbolted.

Main castings and forgings, such as the impeller and casing are radiographed to insure soundness. All pressure containing welds on the pump unit and associated piping are 100% radiographed according to standards of the ASME Pressure Vessel Code. In addition, the pump impeller and the entire inside surface of each pump cover and casing are dye penetrant checked.

A hydrostatic test of each pump is conducted in accordance with the requirements of the Unfired Pressure Vessel Code, Section VIII, 1960. The allowable working pressure of the pumps is 700 psig.

Each pump unit, including the suction elbow is tested on cold water at low pressure and at full speed to determine the head-capacity characteristics, horsepower required, efficiency and NPSH characteristics throughout the operating flow ranges. A full-scale prototype model of the recirculation pump was built and tested in a hot test loop at reactor pressure and temperature. The prototype test experience will give solid assurance that the pump will meet the design specifications with the available NPSH and operate in a leak-tight, trouble-free manner.

1.6.2 Butterfly Valves for Recirculation Loops

Butterfly valves are located at the suction and the discharge of each of the three constant speed recirculation pumps. Full penetration welds are used to weld the butterfly valves into the recirculation piping.

The valves are designed for a line pressure of 700 psig and a maximum differential pressure across the disk of 50 psi. The valve body and disk are made of 304 stainless steel which is fully radiograph inspected. The valve

shaft seals are of the stuffing box type provided with drain lines to the reactor building sump. These drains provide control of any radioactive leakage. The butterfly valves are of the angle seat type to minimize leakage at shut off.

The valve operators are of the electric motor gear drive type which fail in position on loss of operator power. The pump discharge valves, which are used to control flow, have operators so chosen that the total rate of change of flow would not exceed 455 gpm/sec. which corresponds to an external reactivity change of approximately 5¢/sec.

The discharge and suction valves are limited in closure to 5° from the valve seat during reactor operation thereby allowing a continual mixing of water in the down loop(s) to ensure uniform temperature. A pump is prevented from starting until its discharge valve is 5° from its valve seat, to allow a gradual increase in flow upon starting. Also the discharge valve automatically opens to 40° from the valve seat when a pump is started so that operating time in the cavitation region is minimized. Interlocks with the discharge valves prevent a second or third pump from starting if any discharge valve is opening, in order that the rate of flow addition does not exceed 455 gpm/sec. This rate of flow addition is not exceeded upon startup of a pump or during gang operation of the three discharge valves in the control range.

In order to prevent the external reactivity change from exceeding 5¢/sec, the reactor control rods are interlocked with the pump discharge valves

to prevent control rod withdrawal when a discharge valve is opening. Also opening of any discharge valve is prevented by interlocks if a control rod is being withdrawn.

During shutdown valves located at the pump discharge and suction can be closed to permit maintenance.

1.6.3 Recirculation Piping

The recirculation piping is fabricated in accordance with the ASA B31.1 1955 Code for Pressure Piping and any applicable nuclear case rulings. Both the inlet and outlet piping from the three recirculation pumps is 22 inches O.D. pipe. From the vessel to a point 12 inches from the butterfly valves the piping is 5/8 inch thick ASTM-A264 carbon steel backing material clad with 1/8 inch thick stainless steel type 304L. From a point 12 inches before the butterfly valves to the pump suction elbow the piping is 13/16 inch thick type 304 stainless steel. To match the pump casing the pump suction elbow is 1 inch thick type 304 stainless steel. A typical pump loop showing both the dimensions and the isometric line diagram is illustrated in Figures 1.23 and 1.24. All piping is hydrostatically tested in accordance with ASTM-A155.

The fabrication of the pipe is rigidly controlled through process specifications prepared by Allis-Chalmers Mfg. Company. All welds are radiographed in accordance with the ASME Unfired Pressure Vessel Code paragraph UW-51 in addition to other non-destructive tests. Each welder must be qualified prior to doing any welding on the pipe. The pipe is carefully machined to proper length in the fabrication shops and then erected in the field with care

and precision to minimize cold stresses in the piping. The stainless steel butterfly valve body and its assembled end pieces form the final fit-up piece in each pipe. The distance allotted this fit-up piece in the pipe run is carefully measured in the field and the fit-up piece is machined to fit allowing for weld shrinkage. The field welds at each end of the fit-up piece are the final welds in each pipe run.

The recirculation piping was analyzed to determine the forces and moments acting on the reactor vessel and the recirculation pumps and the resulting stresses. Results were obtained using the Kellogg General Analytical Method, with each loop considered as a two anchor-point system having one intermediate restraint. The connection of the suction and discharge pipes at the vessel were the two anchor points, and the pump the intermediate restraint. This method was adapted for use in the piping flexibility program coded for the IBM-704 computer. In addition, a final piping analysis was performed by the Arthur D. Little Company as a check of the Allis-Chalmers analysis.

The results of the stress analysis showing the maximum stress anticipated in the pipe loop is shown in Table 1.6. This data includes stresses resulting from forces and moments due to the thermal expansion of all the recirculation loop components. The forces and moments as calculated are shown in Table 1.7. It should be noted that the maximum stresses are well within code allowances.

The maximum stress calculations are based on an assumption that the joint between the stainless steel and the clad pipe is a sharp, butt joint. The actual joint design is shown in Figure 1.25. The transition from stainless

TABLE 1.6

CLASSIFICATION OF STRESS	AT 1.75 INCHES FROM DISSIMILAR METAL JOINT		AT DISSIMILAR* METAL JOINT		ALLOWABLE STRESS
	(1)	(2)	(1)	(2)	
Thermal Expansion Stresses					SALL.
Clad Carbon Pipe	+12517 (long.)	+11510 (4)	+17219 (Tang.)	+15100 (4)	23625
Austenitic Stainless Pipe	+ 6546 (long.)	+ 9950 (3)	+10915 (Tang.)	+11090 (3)	26563
Mechanical Stresses					Sh
Clad Carbon Pipe			+ 6529 (long.)		15750
Austenitic Stainless Pipe			+ 5504 (long.)		12500
Mechanical Plus Thermal Stresses					Syield @490°F
Clad Carbon Pipe	+21541 (Tang.)	19650 (4)	+29119 (Tang.)	18300 (3) 25420 (4) 16200 (4) (includes radial stress)	31000 psi
Austenitic Stainless Pipe	+12050 (long.)	9389 (3) 13700 (includes radial stress)	+ 5698 (long.)	6626 (3)	19500 psi

Where SALL = $f(1.25 S_c + 0.25 S_h)$

S_c = Allowable stress in cold condition (ambient temperature)

S_h = Allowable stress in hot condition (operation temperature at 490°F)

Syield = yield strength of material at operating temperature.

*The stresses due to the dissimilar metal joint are included in these figures based on a butt joint of two dissimilar metals. However the joint design calls for a gradual metal transition resulting in a considerable reduction in stresses.

1 = Max. Principal Stress

2 = Equivalent uniaxial intensity of combined stress

3 = Based on Maximum Shear Theory (preferred by ASME)

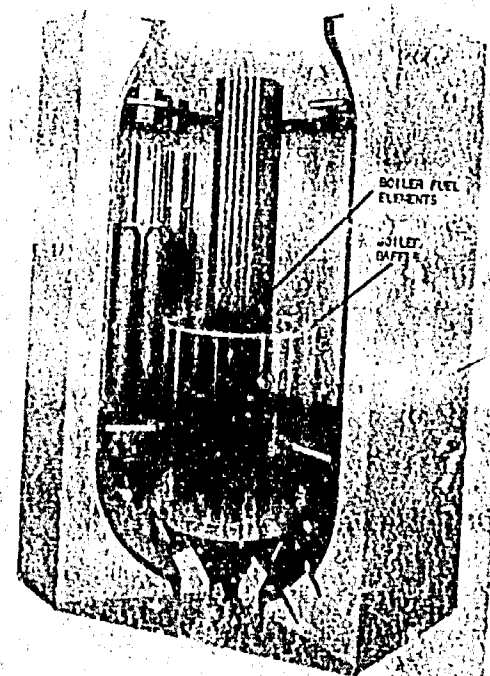
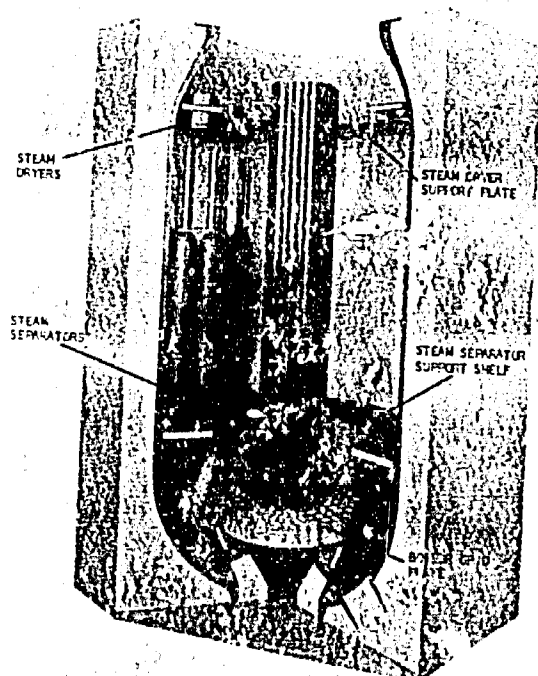
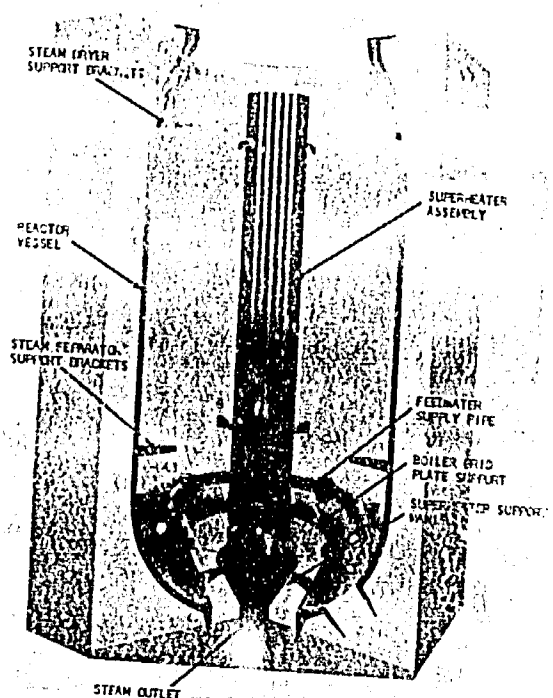
4 = Based on Maximum Distortion - Energy Theory

TABLE 1.7

ACTION OF PIPES ON PUMP & VESSEL											
FORCE IN SUCTION PIPE - LBS AT PUMP (POINT 4) AT VESSEL (POINT 0')						FORCE IN DISCHARGE PIPE - LBS AT PUMP (POINT 7) AT VESSEL (POINT 9)					
F _x	F _y	F _z	F _x	F _y	F _z	F _x	F _y	F _z	F _x	F _y	F _z
+2,205	- 58	+ 462	-2,205	+ 58	- 462	+2,597	- 42	+ 936	-2,597	+ 42	- 936
-2,050	+ 357	- 397	+2,050	- 357	+ 397	+7,033	- 1,451	+ 794	-7,033	+ 1,451	- 798
-2,656	+ 348	- 170	+2,656	- 348	+ 170	+7,635	- 943	- 304	-7,635	+ 943	+ 304
MOMENTS IN SUCTION PIPE - FT. LBS. AT PUMP (POINT 4) AT VESSEL (POINT 0')						MOMENTS IN DISCHARGE PIPE - FT. LBS. AT PUMP (POINT 7) AT VESSEL (POINT 9)					
M _x	M _y	M _z	M _x	M _y	M _z	M _x	M _y	M _z	M _x	M _y	M _z
- 382	+2,088	+ 340	+2,422	- 874	- 9,950	- 975	+11,381	-3,603	+ 856	- 1,710	+ 4,370
-1,529	-6,234	-6,967	- 638	-3,550	+ 9,369	+ 0	+ 6,999	+6,005	- 0	+ 7,037	+19,659
- 837	-4,562	-6,151	- 522	-2,972	+11,936	+1,663	- 1,371	+9,712	+ 936	+16,640	+ 8,221

steel type 304 pipe to the clad carbon steel pipe is very gradual. This is accomplished by machining a 15° taper on the end of the clad carbon steel pipe. The pipe is overlaid with stainless steel for the length of the taper.

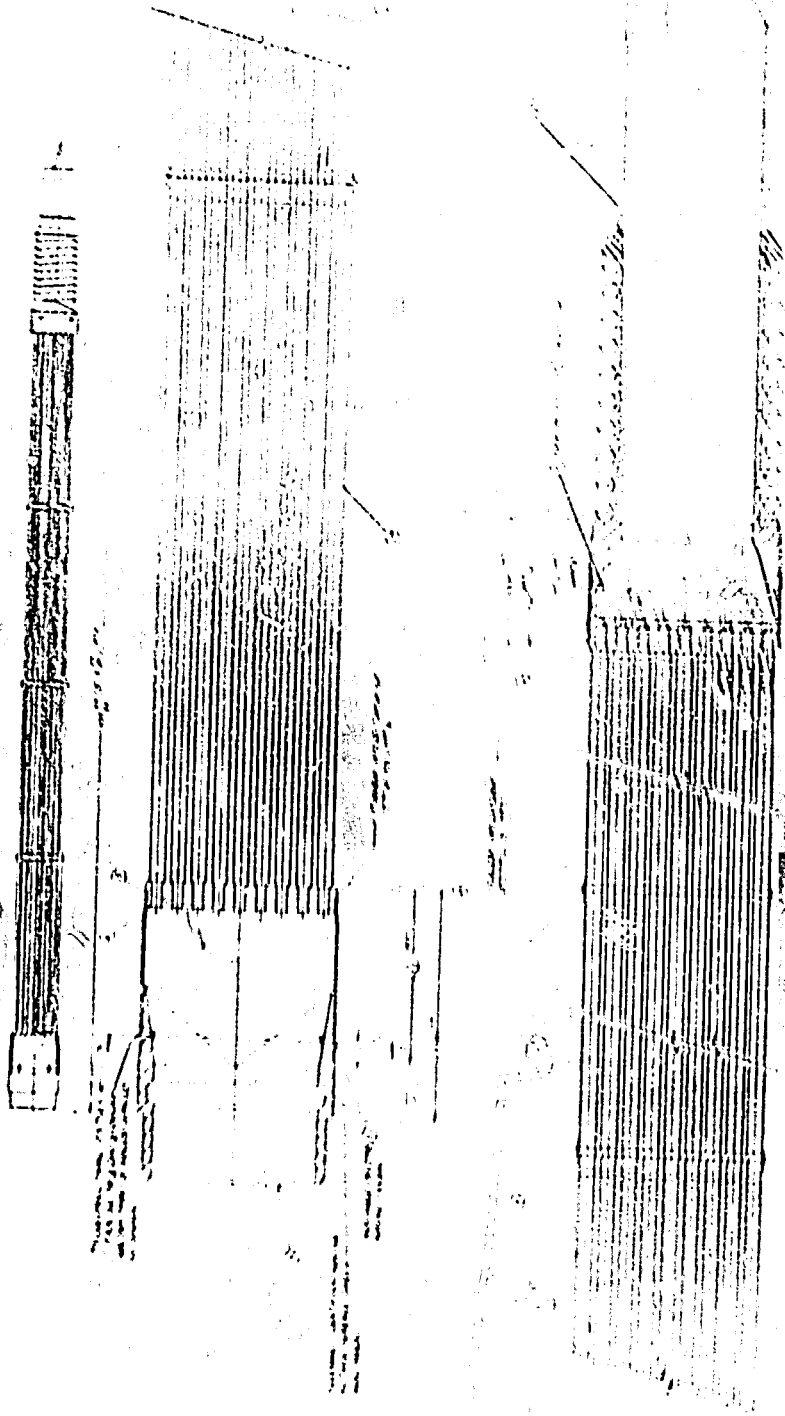
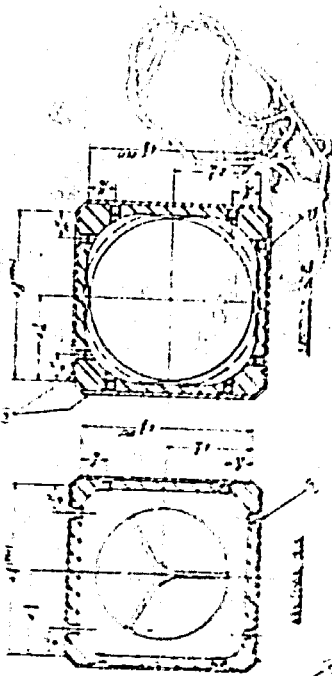
A vibration analysis of the recirculation piping has indicated that the natural frequency of most of the piping is at least twice that corresponding to the normal operating speed of the pumps. On the basis of complete freedom of motion in the horizontal plane some of the pipe frequencies are close to the pump speed. The calculations on which these results are based are extremely simplified and do not account for such factors as elbows, actual loop configuration and static friction at the pump supports. Provision has been made in the pump casing so that a key may be added along the pump centerline to raise the natural frequency well above the pump speed if this should prove to be necessary.



ONE-FOURTH SCALE REACTOR MODEL SHOWING CUTAWAY SECTION

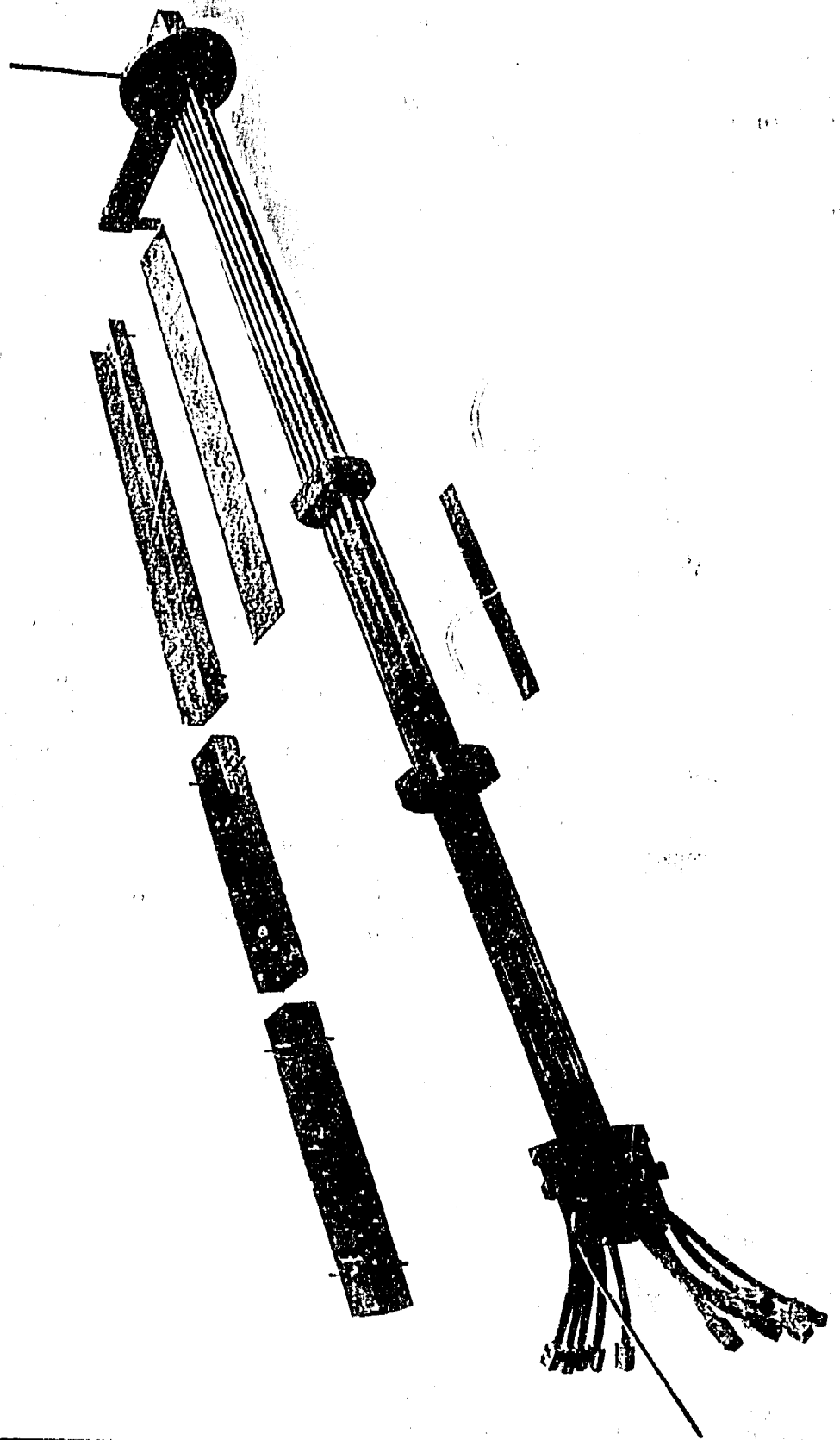
FIG. 1.1
August 15, 1960

1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32	33	34	35	36	37	38	39	40	41	42	43	44	45	46	47	48	49	50	51	52	53	54	55	56	57	58	59	60	61	62	63	64	65	66	67	68	69	70	71	72	73	74	75	76	77	78	79	80	81	82	83	84	85	86	87	88	89	90	91	92	93	94	95	96	97	98	99	100
1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32	33	34	35	36	37	38	39	40	41	42	43	44	45	46	47	48	49	50	51	52	53	54	55	56	57	58	59	60	61	62	63	64	65	66	67	68	69	70	71	72	73	74	75	76	77	78	79	80	81	82	83	84	85	86	87	88	89	90	91	92	93	94	95	96	97	98	99	100



STEP-TUBE PULSED FUEL ELEMENT ASSEMBLY

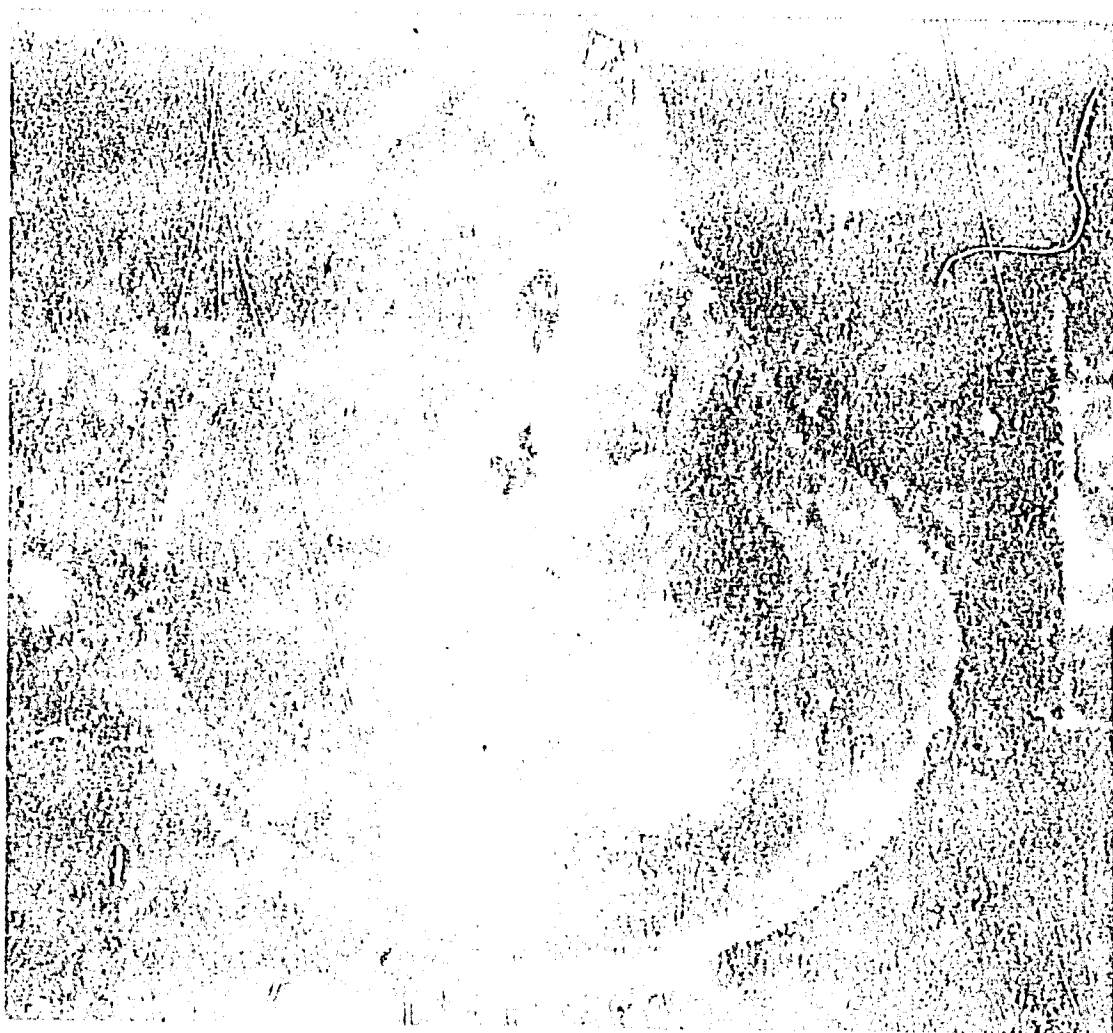
FIG. 1
1966-1970

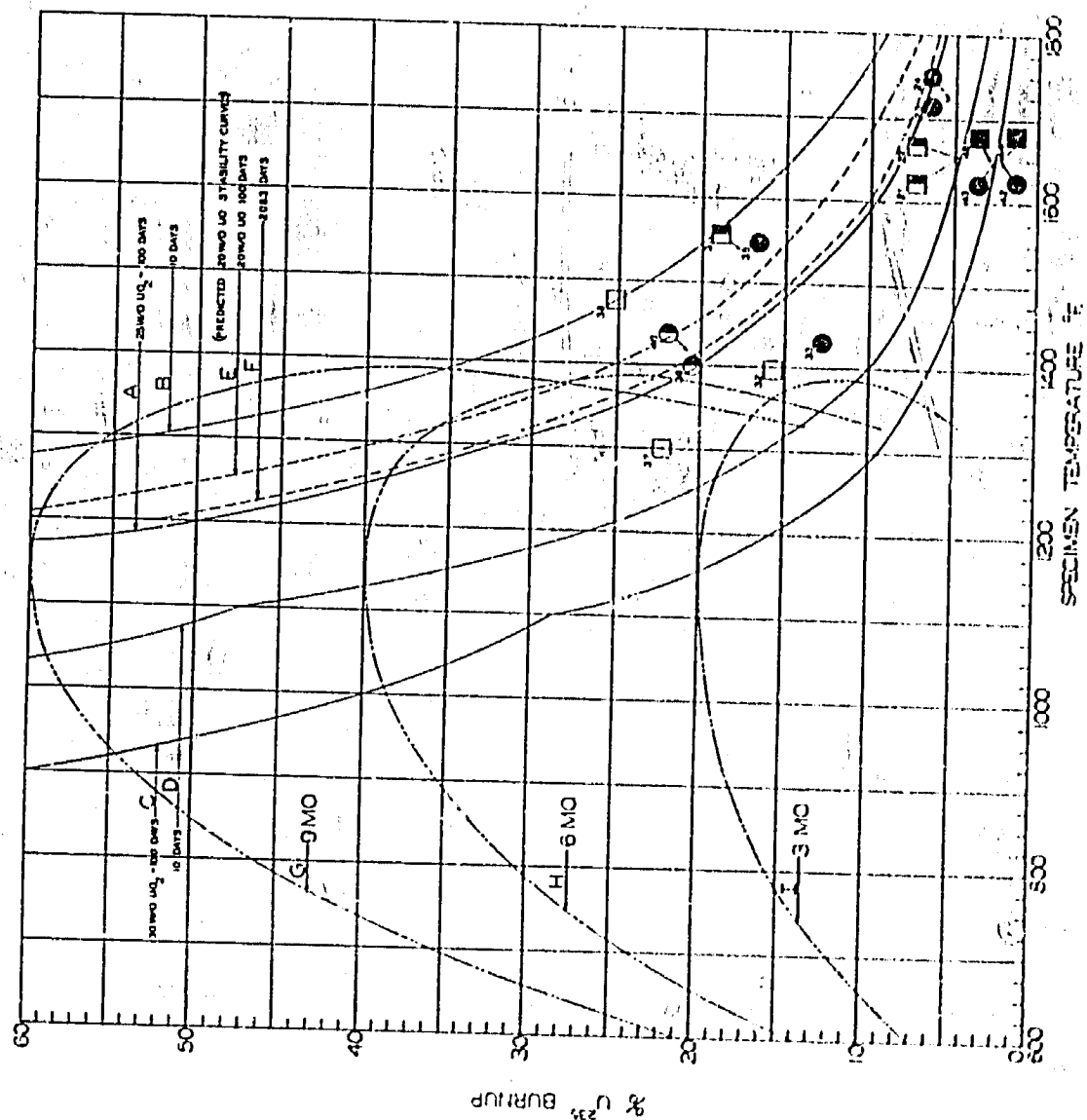


10-8-6 PIN TEST SECTION FOR HEAT TRANSFER LOOP

FIG. 1.3

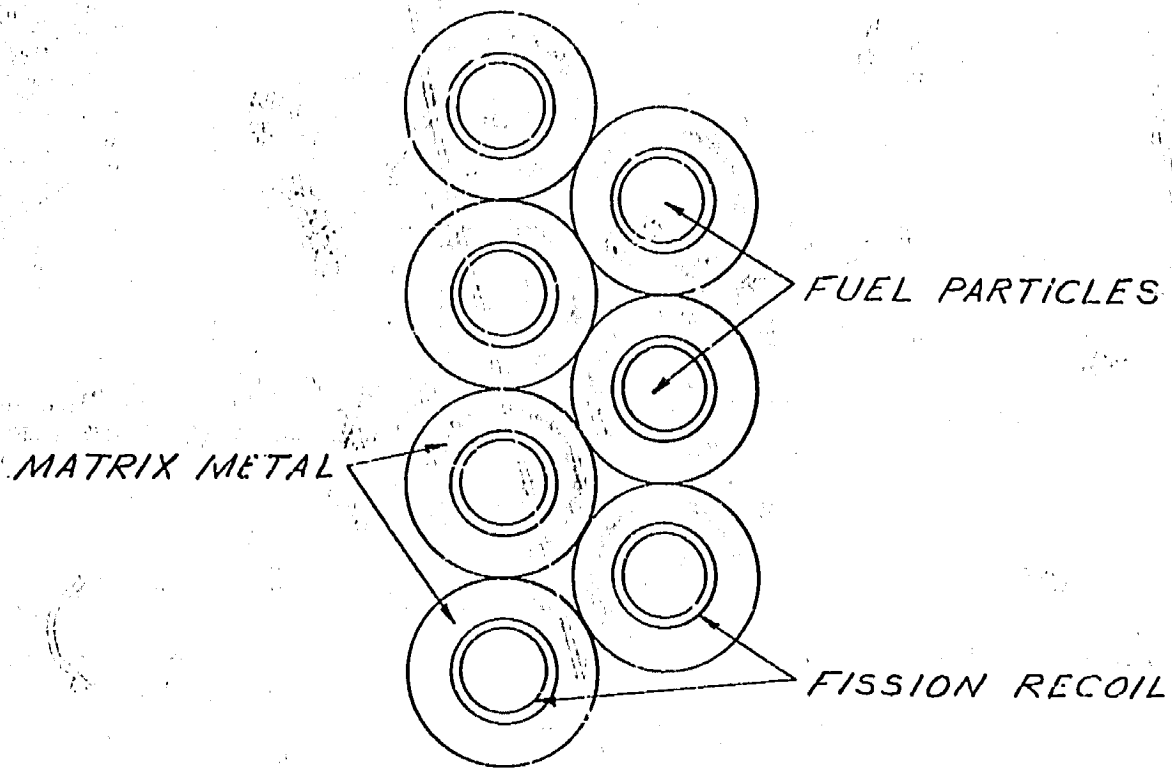
August 15, 1960





UO₂-STAINLESS STEEL IRRADIATION STABILITY (

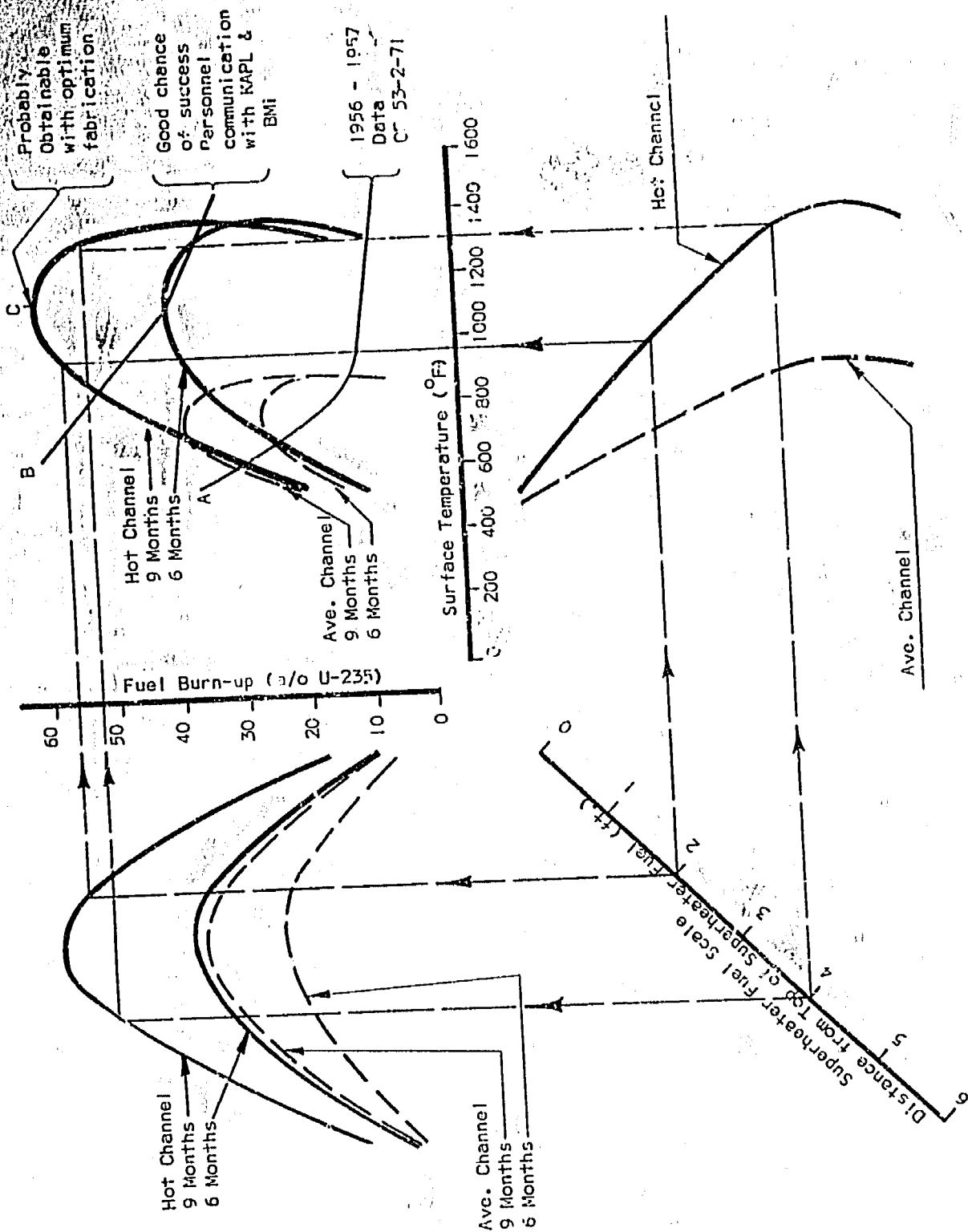
FIG. 1.6
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CERMET FUEL CELL MODEL

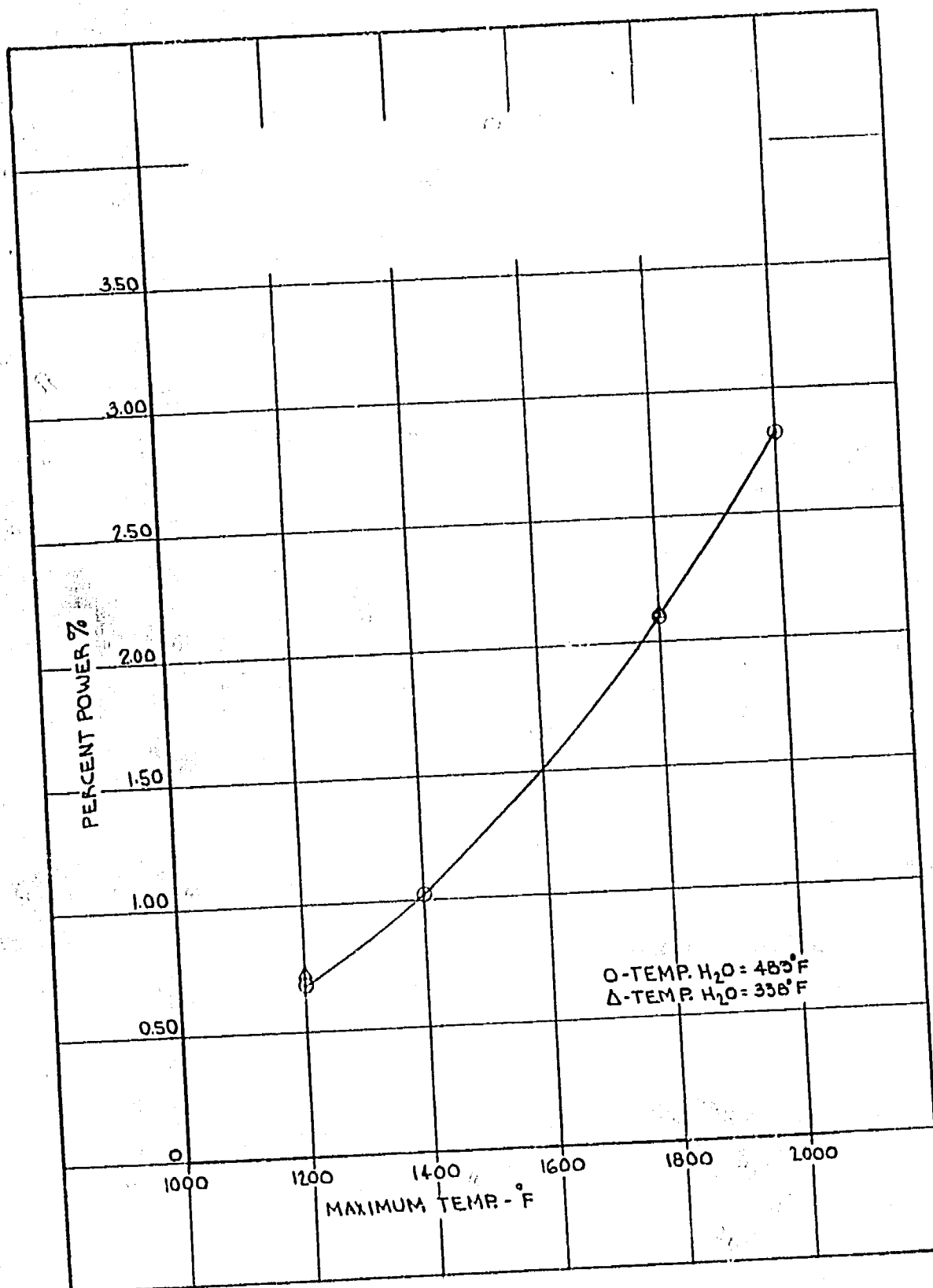
FIG. 1.7

August 15, 1960



CONSTRUCTION OF BURNUP CURVE

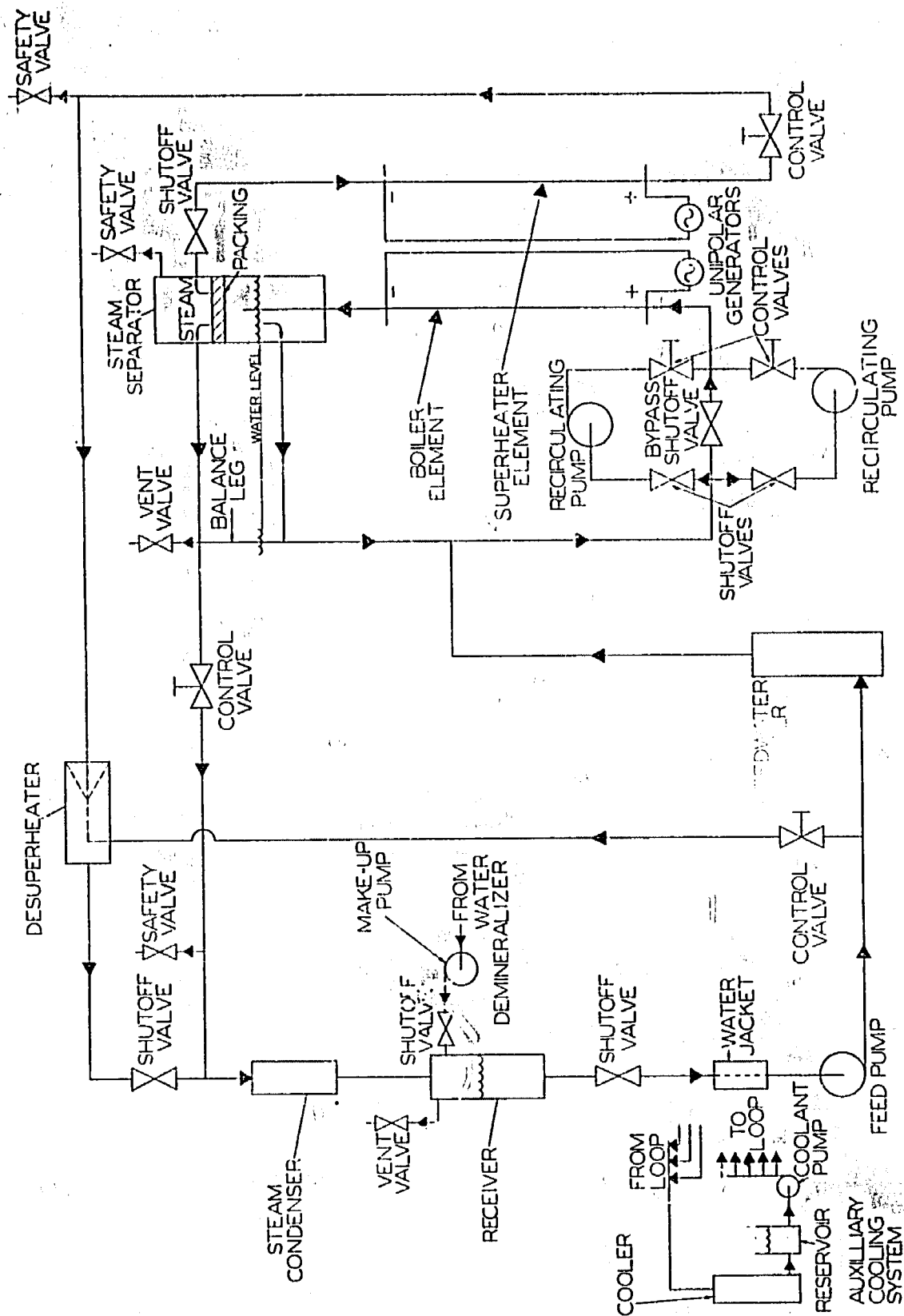
FIG. 1.8
August 15, 1960



ABILITY OF SUPERHEATER TO DISSIPATE HEAT AT NO FLOW CONDITIONS
AND VARIOUS MAXIMUM FUEL ELEMENT SURFACE TEMPERATURES FOR
TWO VALUES OF THE MODERATOR TEMPERATURE

FIG. 1.9

August 15, 1960



SCHEMATIC OF HEAT TRANSFER LOOP

FIG. 1.10
August 15, 1960

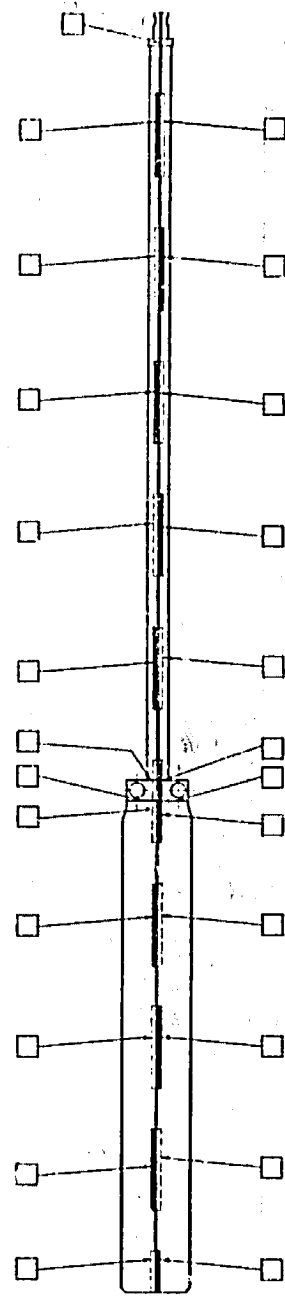
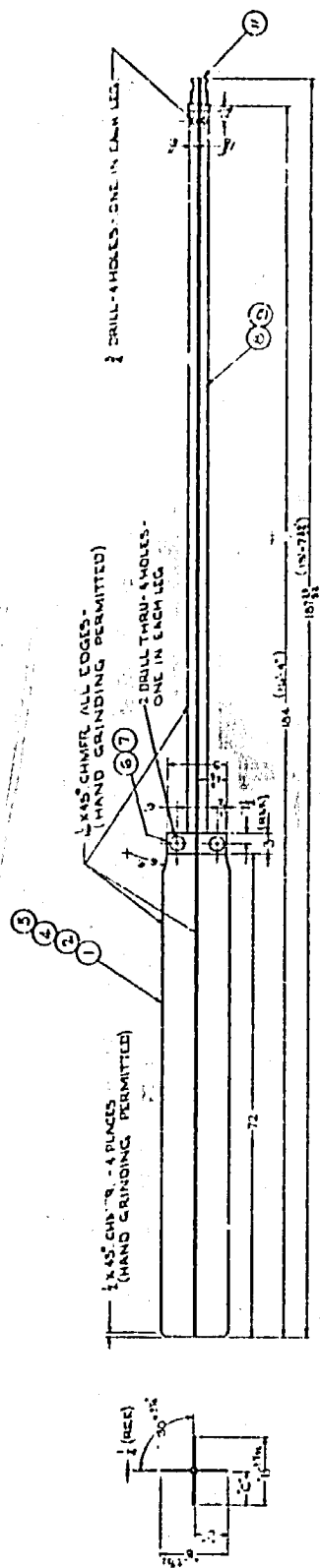
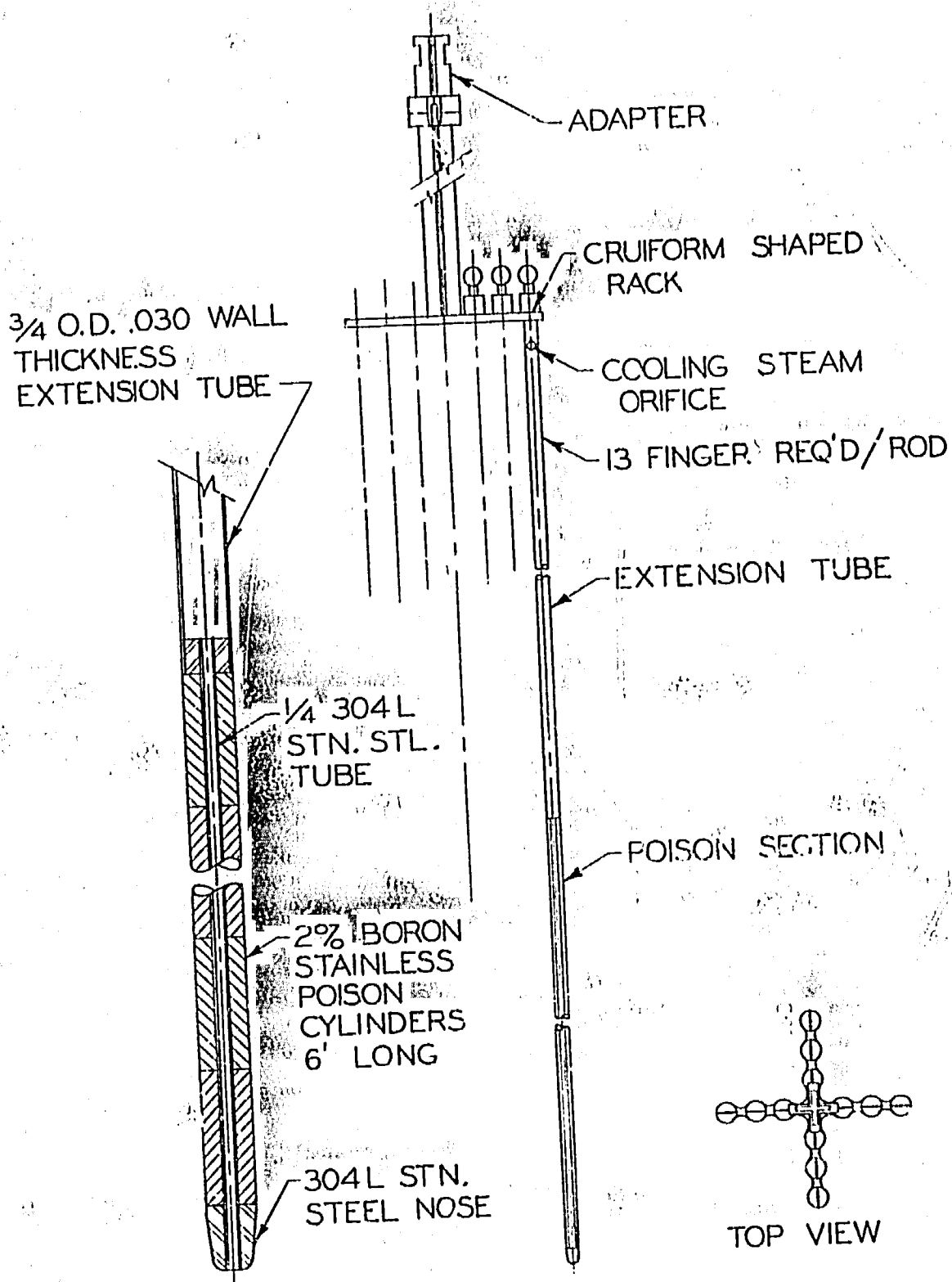


DIAGRAM A

BOILER CONTROL ROD

FIG. 1.11

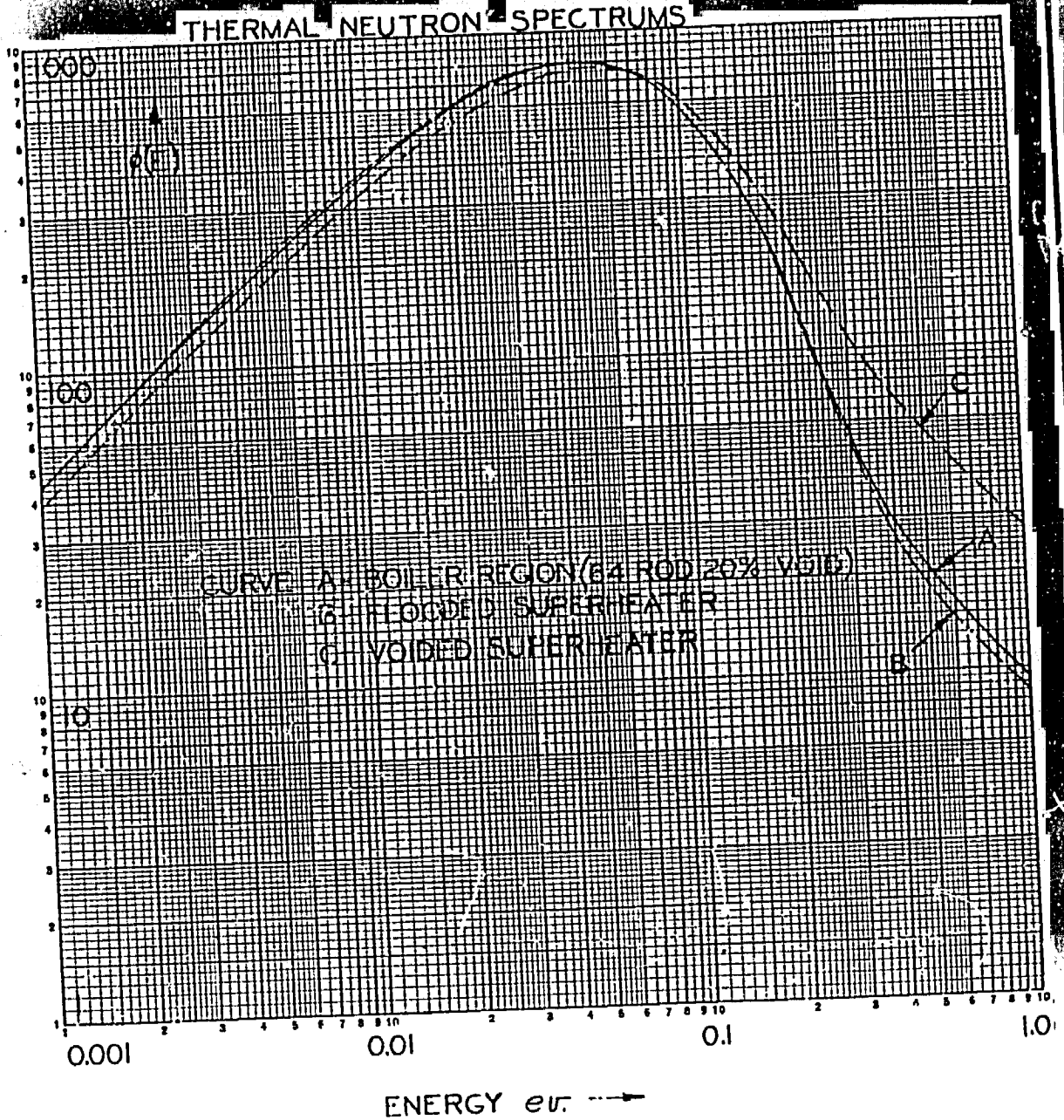
August 15, 1960

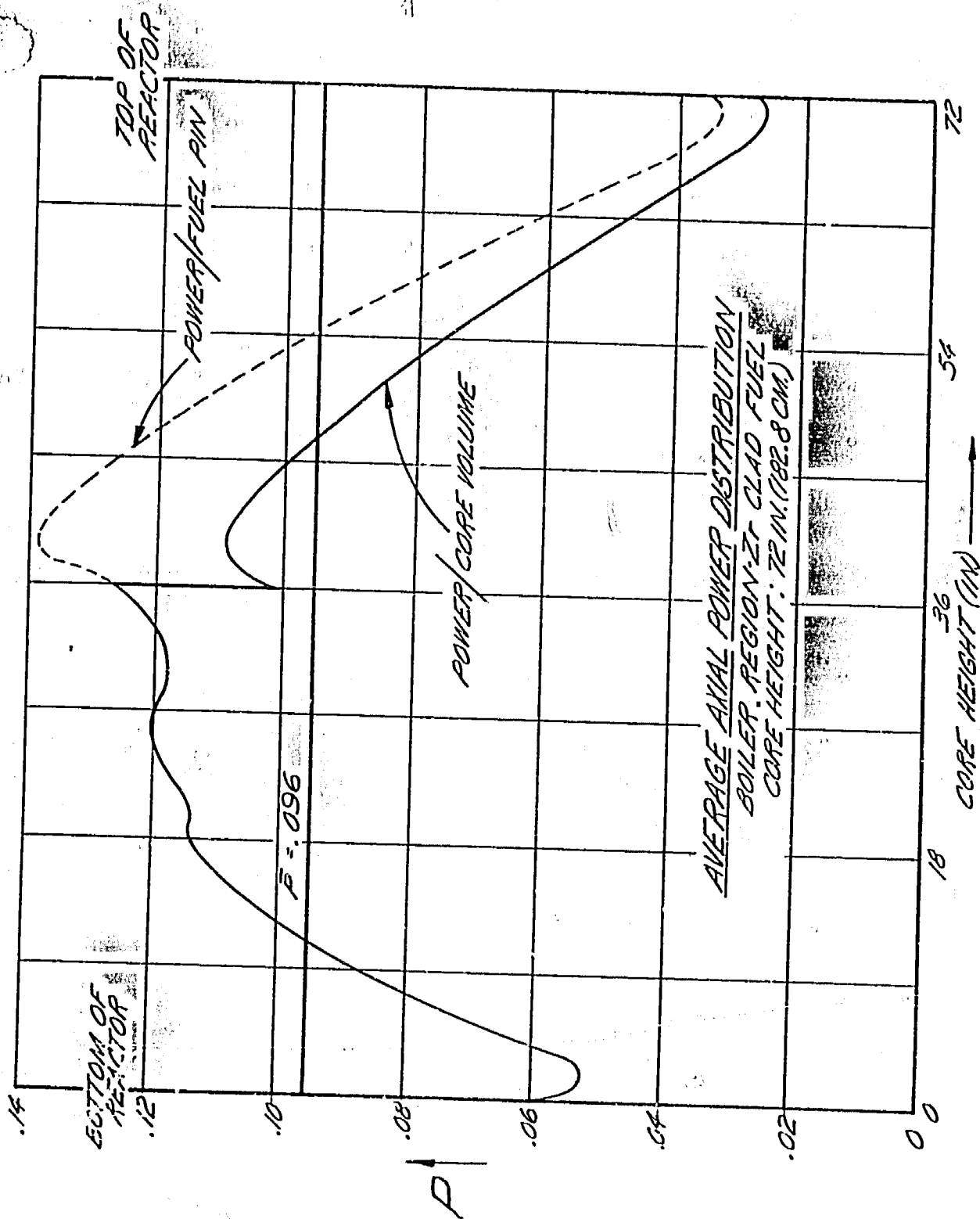


SUPERHEATER CONTROL ROD

FIG. 1.12
August 15, 1960

THERMAL NEUTRON SPECTRUMS

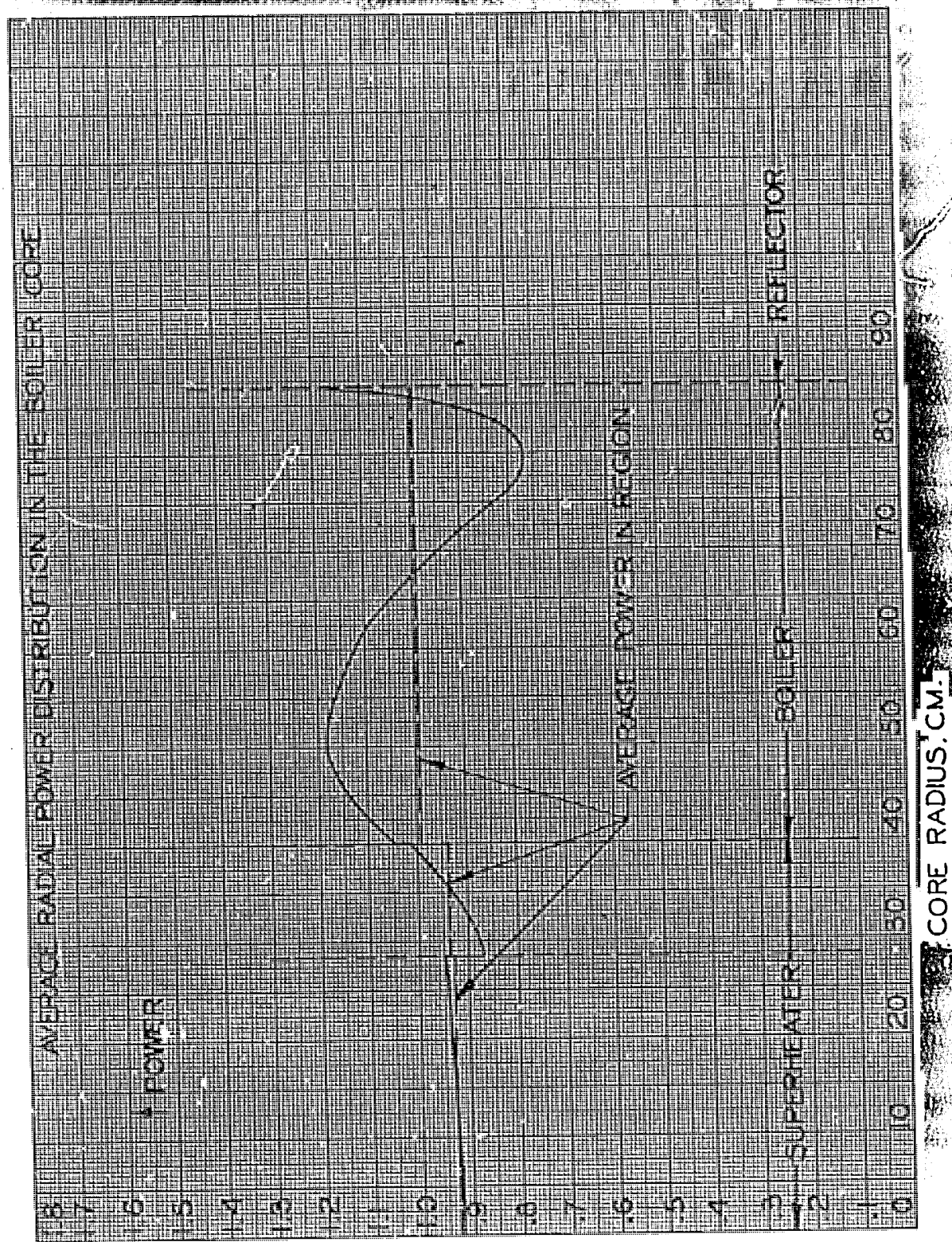




AVERAGE AXIAL POWER DISTRIBUTION

FIG. 1.14

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AVERAGE RADIAL POWER DISTRIBUTION IN THE CORE

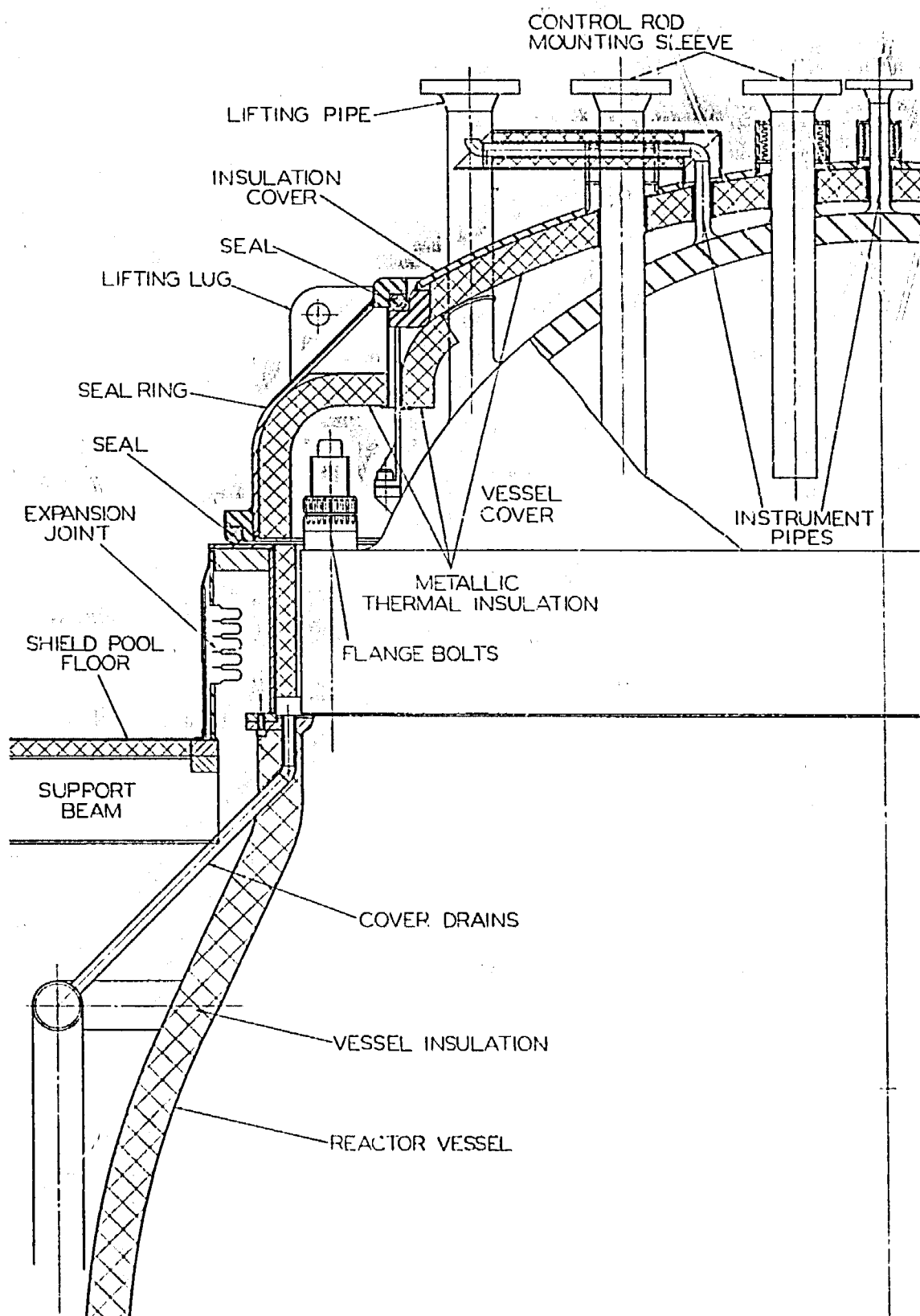
FIG. 1.15
August 15, 1960

OVERSIZE

DOCUMENT IS

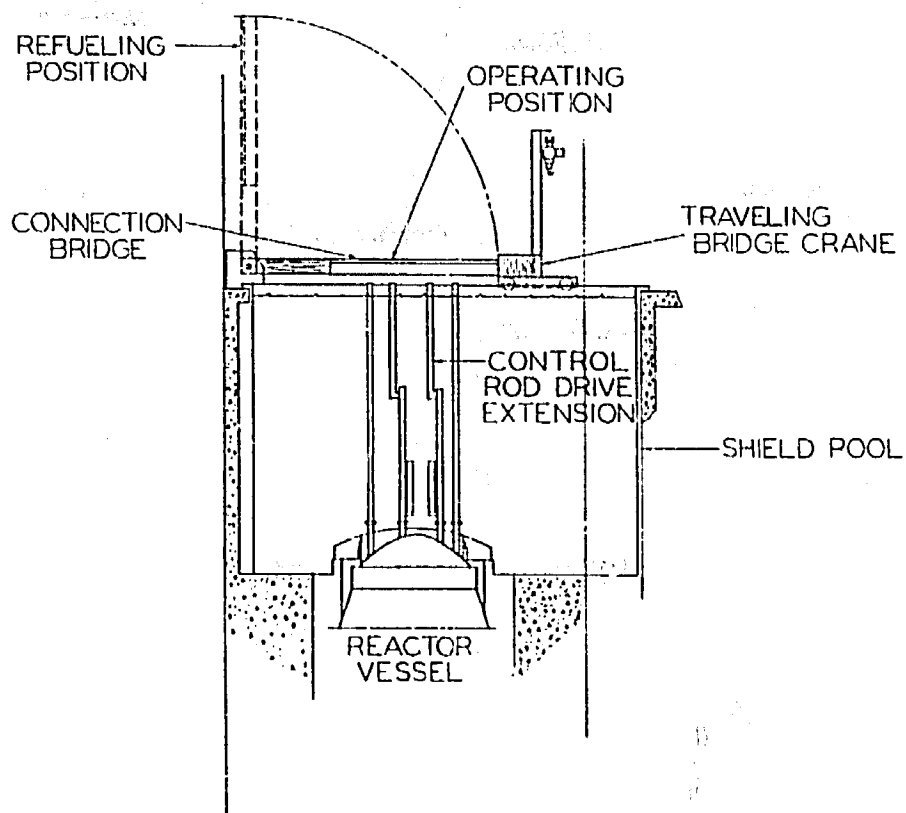
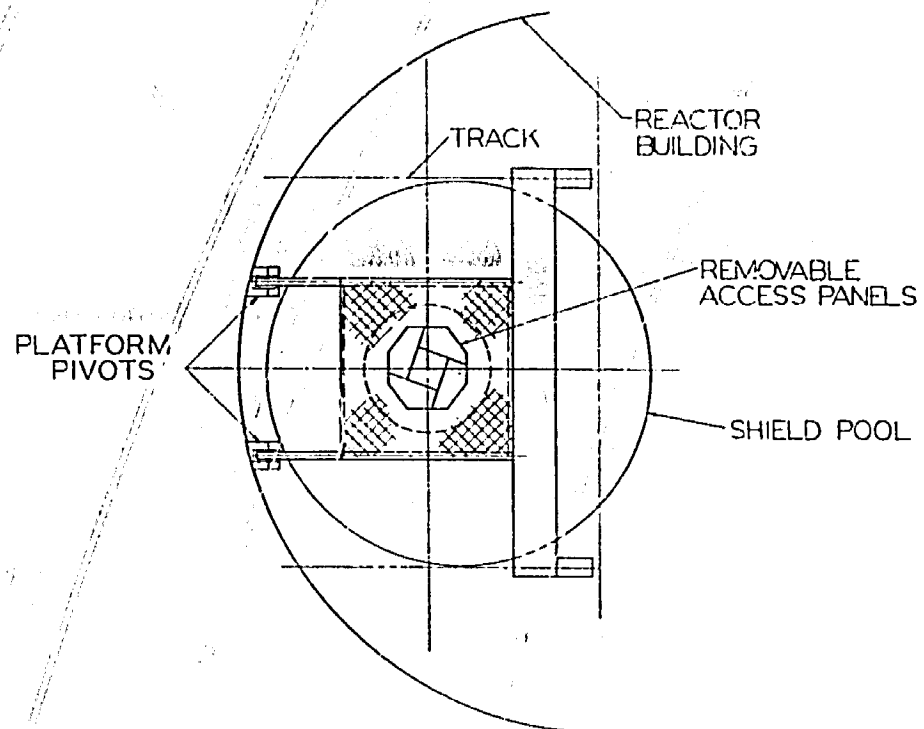
SUPPLEMENTARY

TO MICROFILM FILE



SHIELD POOL SEAL ARRANGEMENT

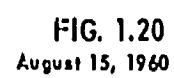
FIG. 1.17
August 15, 1960

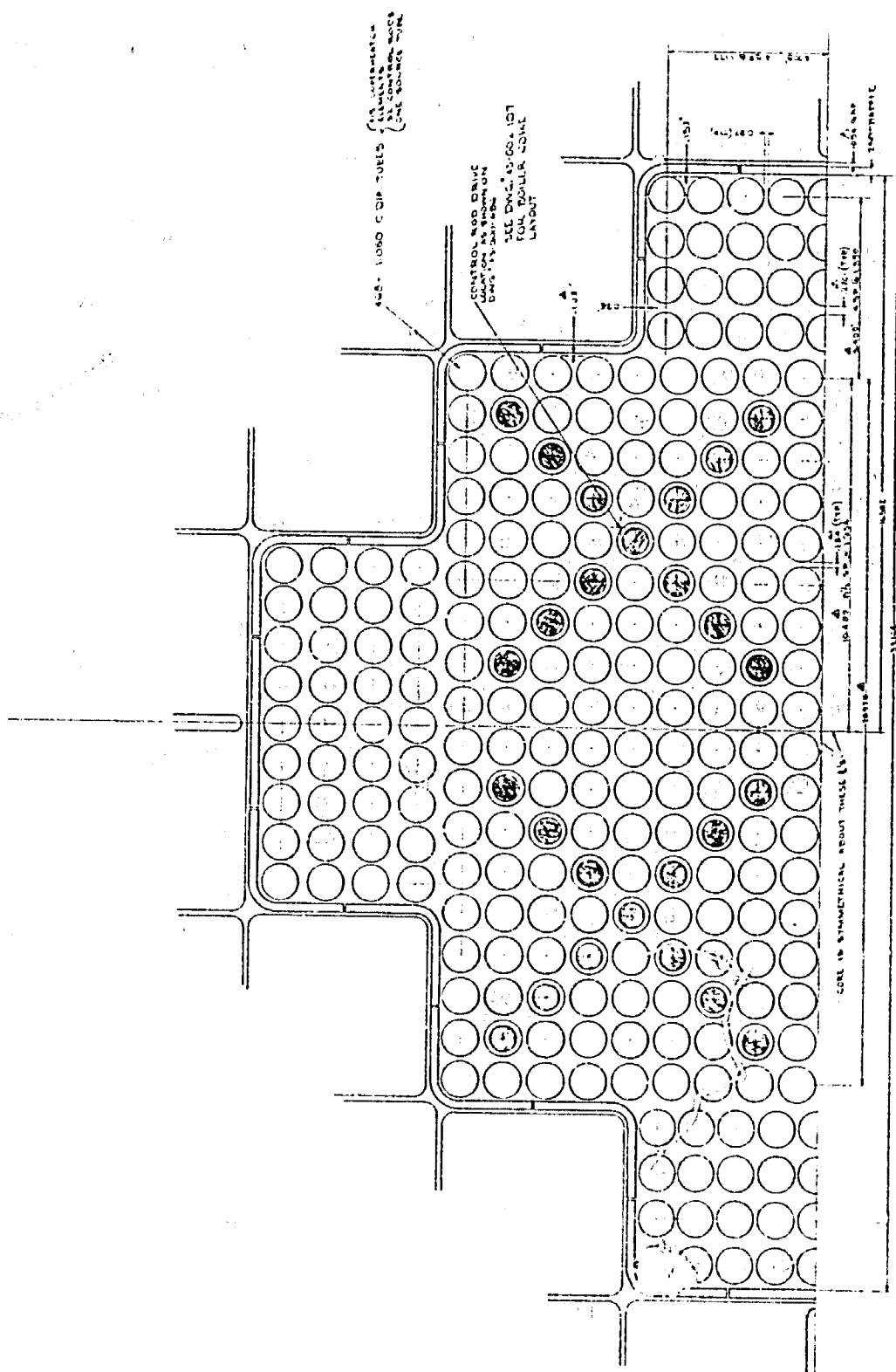


CONTROL ROD DRIVE CONNECTION BRIDGE

FIG. 1.19

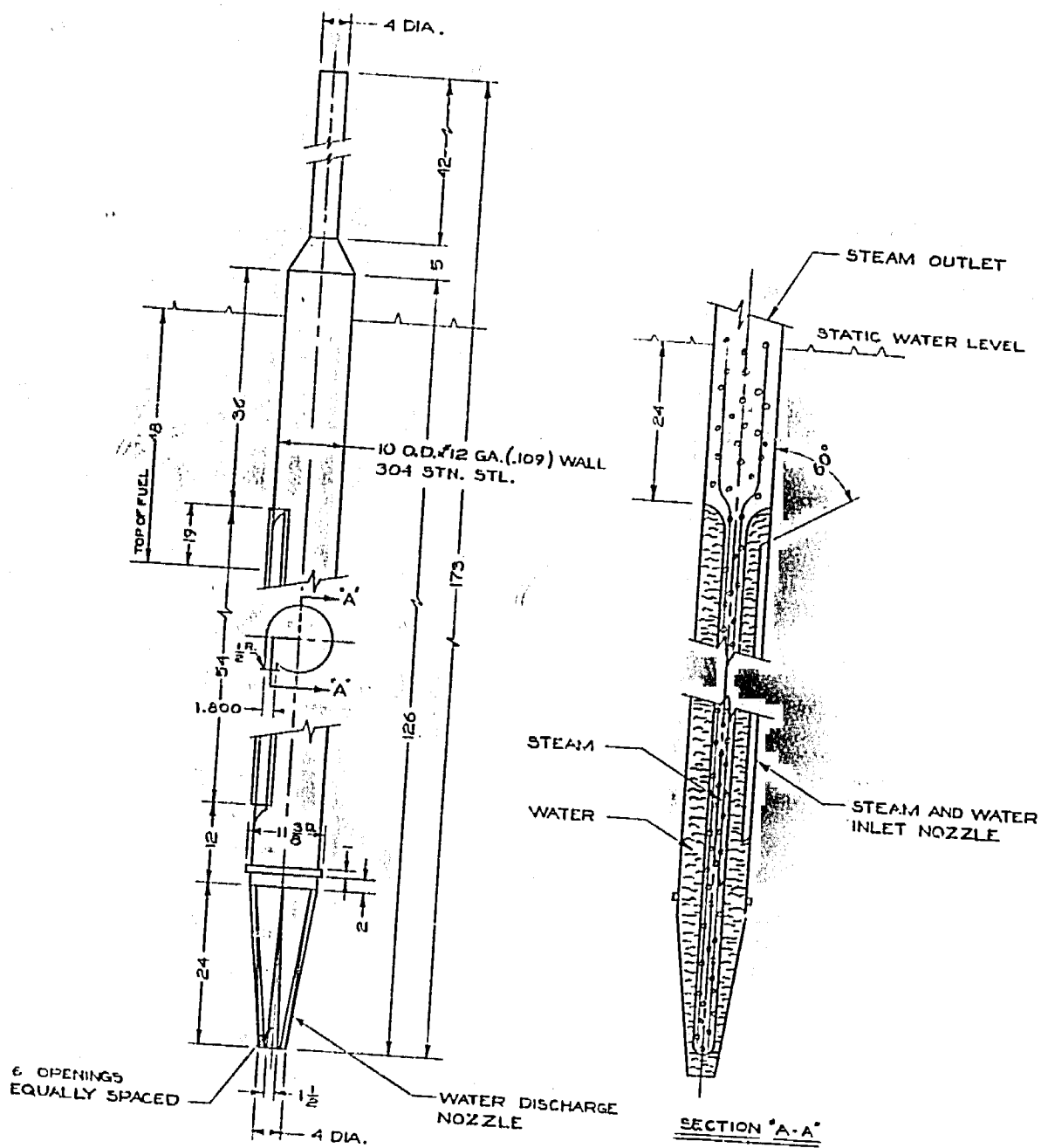
August 15, 1960





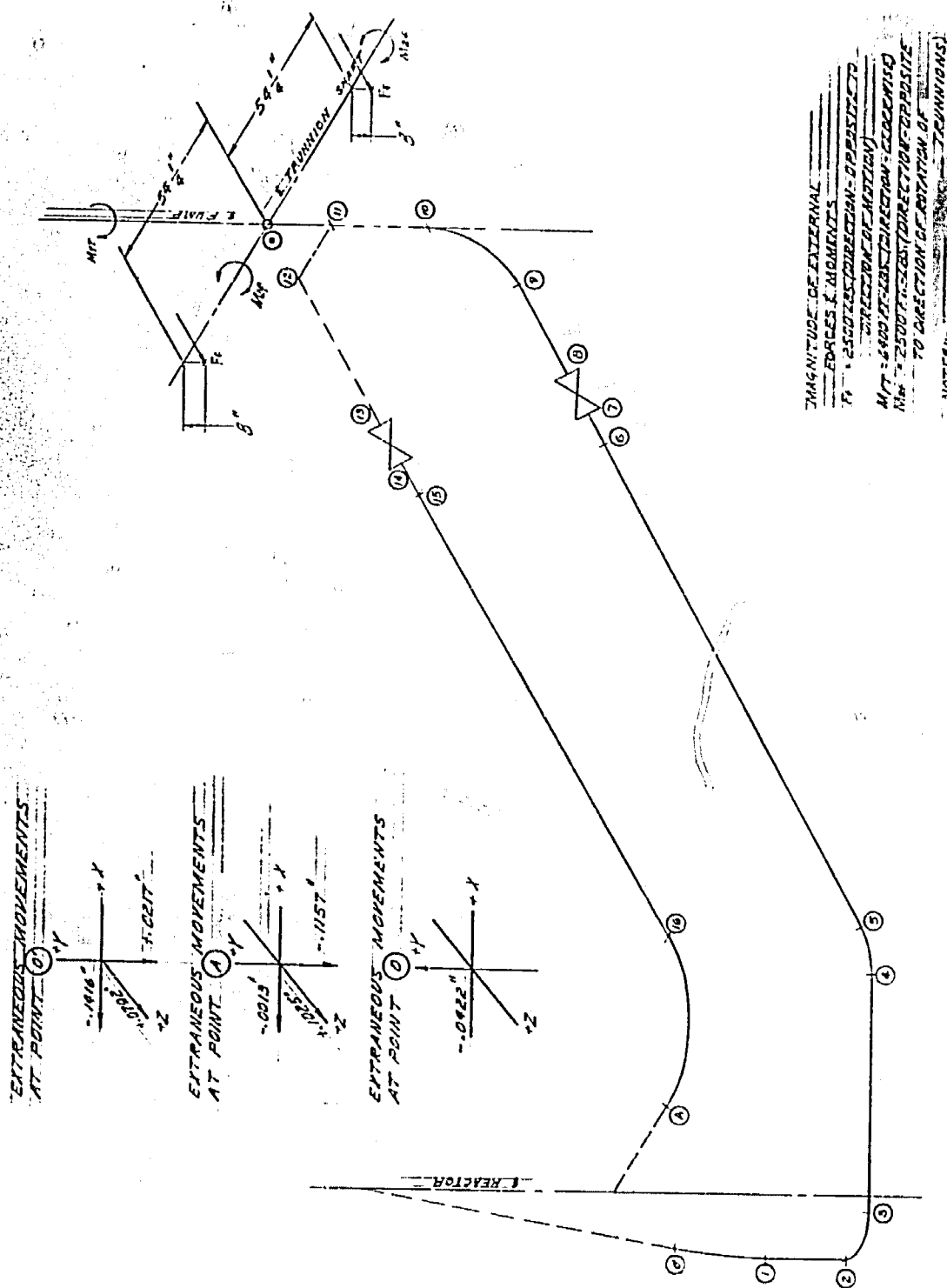
SUPERHEATER CORE LAYOUT

FIG. 1.21
August 15, 1960



10- INCH O.D. STEAM SEPARATOR

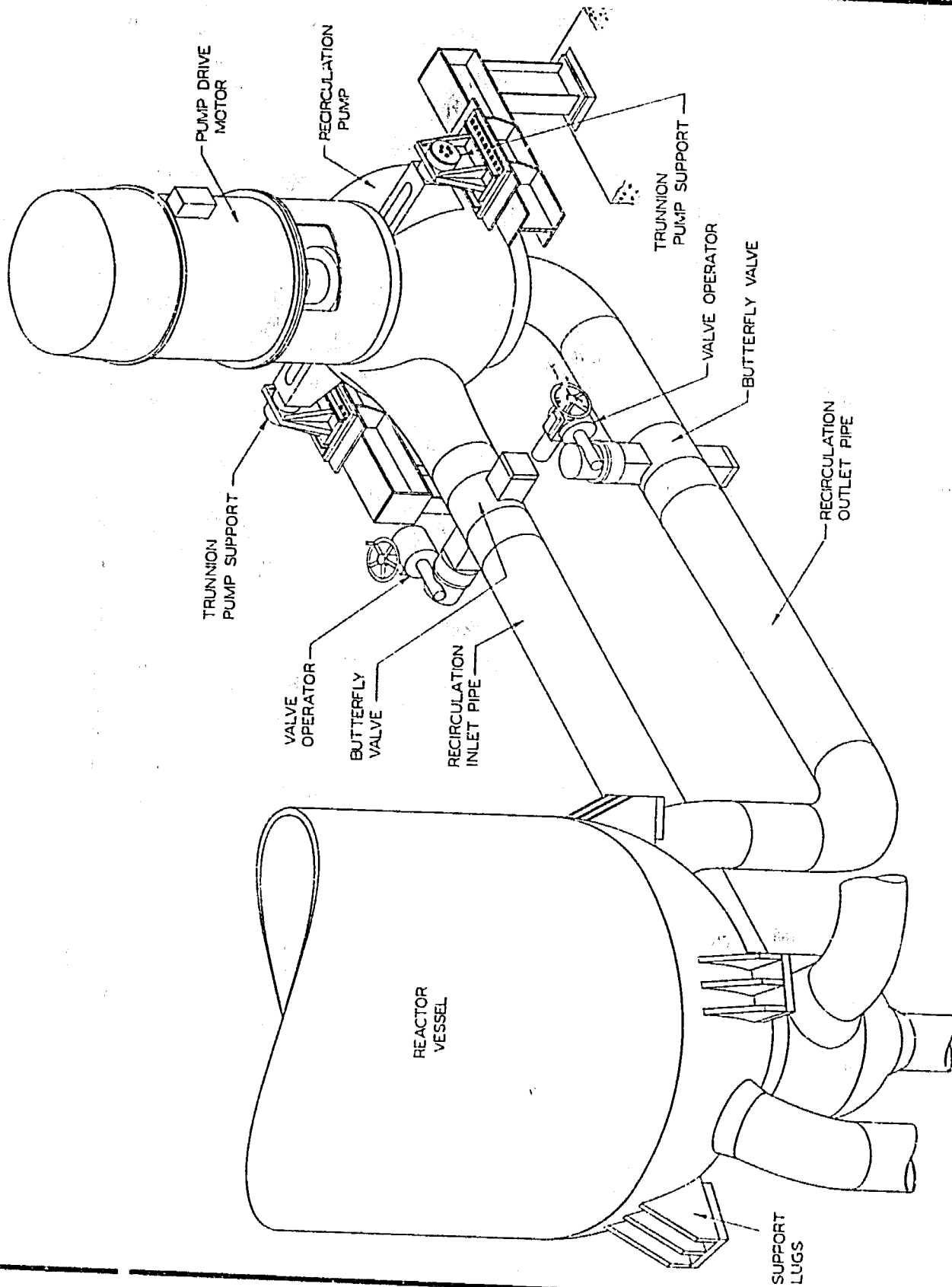
FIG. 1.22
August 15, 1960



RECIRCULATION PIPE LOOP NO. 1

FIG. 1.23

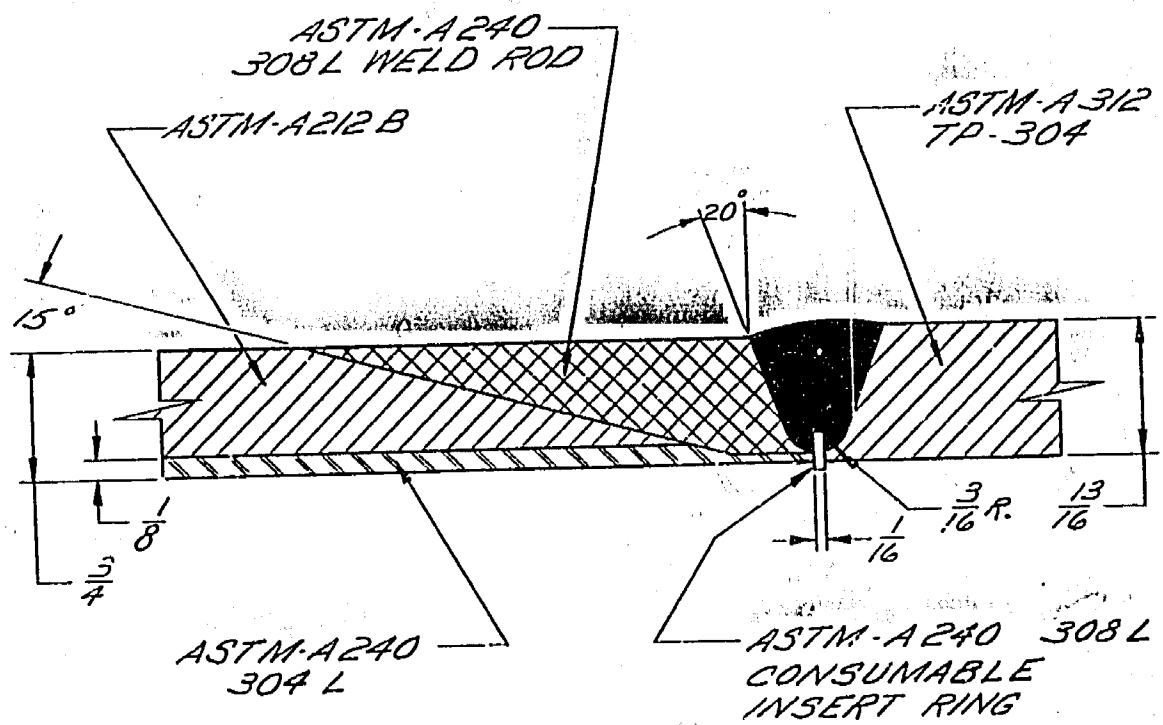
August 15, 1960



RECIRCULATION PUMP LOOP NO. 2

FIG. 1.24

August 15, 1960



WELD PREPARATION OF AUSTENITIC STAINLESS STEEL CLAD
CARBON STEEL PIPE TO AUSTENITIC STAINLESS STEEL PIPE

FIG. 1.25
August 15, 1960

Section 3

REACTOR BUILDING

3.1 Structure

The entire reactor complex will be enclosed in a cylindrical reactor building. The inside diameter is 50 feet. The 120 foot-6 inch inside height includes 83 feet for the cylindrical portion, 25 feet for the hemispherical top and 12 feet 6 inches for the ellipsoidal bottom. Approximately 63 feet of the reactor building is below grade. The building is constructed of preformed steel plates welded together to form a gas-tight enclosure.

Personnel will be able to enter the reactor building through the normal and the emergency access air locks. These two access facilities are located approximately at opposite sides of the building at the operating floor level.

Normal access is by means of an air lock (Figure 3.6) which connects the reactor building to the turbine building. The outer shell of this air lock is cylindrical; 10 feet in diameter and 12 feet-3 inches long terminated at each end by flat reinforced bulkheads. At each end is a 3 foot x 7 foot (clear opening) hinged door which is closed against a rubber gasket by a mechanical latching mechanism. The latching mechanism of the two doors is mechanically interlocked so that no one door can be opened unless the other door is closed and latched. The interlocking mechanism contains a release which, when actuated, permits opening both doors at once. This release will normally be kept locked and will be unlocked only when the reactor is shut down. Each door opens in toward the reactor building so that, in the event

of an accident which results in a pressure buildup in the reactor building, the pressure will push the door against the seal. The latching mechanism is geared to a valve that equalizes the pressure on either side of the door before it is opened.

During normal operation, both doors will be kept closed and latched. A person wishing to enter through the air lock would unlatch and open the first door; enter the air lock; close and latch the first door; unlatch and open the second door; leave the air lock; close and latch the second door.

Emergency access or exit will be by means of an air lock (Figure 3.7) which leads from the reactor building to the outside. The outer shell of this air lock is cylindrical, about 5 feet in diameter and 7 feet-2inches long. The ends are flat reinforced bulkheads with 30 inch diameter (clear opening) hinged doors which open to the inside of the building. The seal, latching mechanism, interlock and pressure equalizing valve is the same as described for the normal access air lock except that there is no release to permit opening both doors at once. This air lock is provided with a means whereby the inside door can be closed and latched from the outside.

The principal purpose of this air lock is to provide emergency exit from the reactor building. To facilitate this movement, the interior door will normally be open and the outside door closed. However, a situation may arise where access to the building is not available through the large air lock. Such a situation could be caused by someone leaving the inside door open, by a jamming of the mechanism, etc. In such a case, the inside door of the emergency air lock can be closed and latched from a position outside of the building and the outside door can then be opened by means of a key, to provide entrance to the air lock. The outside door would then be

closed and latched; the inside door unlatched and opened, and access gained to the building.

Both air locks are built of carbon steel SA-212 Grade B, Firebox quality. The design pressure is the same as that used for the design of the reactor building containment shell. They will be shipped to the site and installed in the reactor building shell before the first pressure and leak test is applied to the reactor building.

A circular equipment door, 11 feet in diameter, will facilitate the removal and replacement of all the equipment contained in the building, except the reactor pressure vessel. The door is bolted and sealed to the shell. To permit installation of the reactor pressure vessel, a hole will be cut in the side of the building. The temporary opening will be closed by welding the cut out section back into position.

The reactor building is insulated on the outside down to an elevation 5 feet below ground level. This insulation is covered with a protective weatherproof coating. Below grade, the outside of the building is protected from corrosion with a permanent coating.

The reactor building (a containment vessel) is designed and constructed according to rigid specifications which require conformance to the ASME "Boiler and Pressure Vessel Code" Section II (Material Specification), Section VIII (Unfired Pressure Vessels), Section IX (Welding Qualifications), Code cases 127CN, 1271 N; and 1272N and the Uniform Building Code as published by Pacific Coast Building Officials Conference. The building is constructed of SA-212, Grade B Firebox quality steel, and inasmuch as it will be exposed to outdoor temperatures below -20F, the SA-212 steel conforms to the specifications for SA-300.

The shell of the reactor building accommodates a design wind pressure load of 30 psf and snow loadings of 35 psf on horizontal surfaces and supports the building crane. The reactor building design pressure is 76 psig which is above the peak pressure resulting from a complete release of all primary coolant. To contain this pressure a shell thickness of 11/16 inch is specified for the hemispherical top and 1 5/8 inches for the cylindrical walls and ellipsoidal bottom.

3.2 Reserve Strength

There is inherent in any pressure vessel designed according to the ASME Code, a capacity to absorb an overload as a result of the factors of safety introduced into the design. In case of a steel vessel, a factor of safety of 4 is specified by the ASME Code. This factor is purposely high in order to allow for fatigue, stress concentrations, variation in the dimension, composition and strength of the steel and imperfections in fabrication, as well as for inaccuracy in the prediction of the load. However, with a ductile steel such as SA-212, plastic deformation may be counted upon to relieve most local overstressing due to the causes outlined above. Thus, since fatigue is not a factor in this case, most of the reserve strength of the steel can be counted upon for resisting an overload. In fact, one series of tests have shown that a well designed and carefully fabricated vessel can exhibit static strengths in excess of the nominal ultimate strength of the metal used for its fabrication. Therefore, an overpressure of twice the design pressure would not be expected to cause the steel vessel to fail, and such a steel vessel probably could be safely subjected to three times the design pressure once during its lifetime and not fail.

As shown in Figure 3.2, the foundations for the reactor building rest on rock. In order to transfer the internal loads from the building to the rock without overstressing the building shell, a bed of concrete is poured between the bottom of the shell and the rock.

It will also be noted on Figure 3.2 that most of this building is below ground level. Earth pressure, therefore, will be exerted on the outside of the building shell. Since the steel shell of the reactor building does not have as much strength in compression, it is backed up where necessary below grade, by a concrete wall. This wall is designed to carry all the loads due to external pressures.

All the interior floors, shielding concrete, piping and equipment, except the 25-ton crane, are supported by concrete walls and columns independently of the outer shell. The crane will be supported by the exterior shell.

3.3 Equipment Layout

The reactor is located in the lower part of the reactor building, surrounded by a concrete shield wall (Figures 3.1 and 3.2). The recirculation pumps are located outside of the reactor shield wall, level with the bottom of the reactor. A shield pool approximately 18 feet deep is located over the reactor. Air fueling and reactor maintenance operations will be performed through this pool, and all fuel will be transferred between this pool and an externally located storage pool through a tube and valve as shown in Figure 3.8.

The reactor will be located off center in the reactor building. The recirculation pumps will be arranged as shown in Figures 3.1 and 3.2. Floors

above the recirculation pump rooms will be used for auxiliary equipment and instrumentation. A circular crane will be used to service the equipment. The recirculation pumps will be serviced by a local monorail crane which can move them to a hatchway under the main building crane.

The shielding arrangement of the reactor and the recirculation pumps and piping are basically as shown in Figures 3.1 and 3.2. The reactor vessel is located below the outside grade elevation. Therefore, the surrounding earth can be utilized to minimize the amount of concrete required in one segment of the reactor shield. Reactor building size is also reduced by this approach.

The relative elevation of the equipment is such that the water level in the shield pool located in the reactor building is at the same elevation as the water level of the fuel storage pool located in the fuel handling building. This facilitates the design and operation of the fuel transfer passage between the pools.

The thickness of shielding around other equipment in the reactor building is sufficient to provide access to the following areas while the plant is in operation:

1. The operating floor -- uppermost floor in the reactor building and the floor on which electrical control equipment is located.
2. Non-radioactive areas of the next lower floor, the equipment floor.
3. Pump rooms on the bottom floor (limited access). (Refer to Section 4 for further detail)

Some routine maintenance can be performed while the plant is in operation. The recirculation pump motors and seals can be inspected if necessary. By shutting down a recirculation loop the motor in that loop can be disconnected and removed. The pump impeller, shaft and seal, however, can be removed only when the reactor is shut down and unpressurized. Auxiliary equipment on the equipment floor can be inspected and routine maintenance performed.

All equipment is arranged to provide a minimum of handling difficulties for installation or removal, and to permit sufficient working space for maintenance. Openings are provided in the various floors to allow each piece of equipment to be transferred between the location at which it will operate and the operating floor. From the operating floor all equipment, except the reactor pressure vessel can be removed from the building.

The shield pool is accessible from the operating floor. Those pool operations which are performed while the plant is not in operation, such as the transfer of fuel elements, reactor internals, and control rods between the shield and storage pools, can be performed from the bridge platform located over the pool and accessible from the operating floor. Sufficient space is provided in the shield pool for storage of fuel handling equipment and the pressure vessel lid during reactor fueling or during maintenance operations. The importance of operator freedom for the manipulation of fuel elements and reactor internals through 18 feet of water is recognized. Consequently, sufficient clearance is provided above and around the pool for any foreseeable type of pool operation. This is also true of requirements for servicing the control rod drives.

When not in use, the fuel canal will be closed by a gate valve (Figure 3.8) to prevent flow out of the building during a pressure rise in the reactor building. This valve will always be closed during reactor operation.

Pipes passing through the reactor building shell are grouped together. Sufficient clearances between penetrations are allowed for convenient inspection of the penetration seals.

The majority of piping connections in the reactor building and the turbine building are located below the operating floor level which is 3 feet above grade. This will allow all piping to be extended between the buildings in an underground pipe tunnel.

Electrical leads penetrate the reactor building below grade within a separate tunnel located under the personnel access air lock. Ventilating ducts penetrate the building shell below the operating floor.

All lines which could conceivably pass a dangerous amount of radioactivity in case of an accident are equipped with isolation valves such as, the main steam line (closing time approx. 10 sec.), the reactor water purification line, and the feedwater line. All other lines utilize those stop valves which normally are required to isolate such equipment during non-operating periods as the shield pool, shield pool coolers, and demineralized water storage tank. These particular lines terminate in closed vessels or equipment outside the reactor building and will not constitute a hazard even if they did pass some radioactivity.

The isolation valves will close automatically only when the activity leaving the reactor building reaches a predetermined level or on loss of power and will insure that the pipe line itself is not a source of leakage

from the building. Incoming lines including the feedwater line are equipped with check valves. Outgoing lines have motor or solenoid operated isolation valves. The operating mechanism of the isolation valves are designed to close on power failure. These valves are located on the inside of the building close to the wall.

The penetrations in the reactor building wall are sealed against leakage under internal pressure. Hot lines are sealed by using a double bellows or equivalent type of expansion joints (Figure 3.3). This allows for axial and lateral movement of the pipe due to thermal expansion without imposing large stresses on the joint or equipment to which they are attached. Penetrations of lines carrying fluids near room temperature are sealed by welding each line to the wall of the reactor building, as shown in Figure 3.4. Ventilating air ducts are sufficiently large to require that a sleeve be provided as part of the reactor building shell to which the duct is welded.

Depending upon the type and individual requirements, electrical leads pass through the building wall in sealed pipe sleeves or through a double wall section of the building. The volume formed by the double wall is filled with an epoxy resin. A gland type compression fitting is used to seal around solid core mineral insulated copper-jacketed cables (Figure 3.5). Where Neoprene, Polyvinyl Chloride or lead covered cables are used, an end chamber and sealing bushing is screwed onto the penetrating pipe. The end chamber is provided with a filling plug so that the internal chamber surrounding the cable can be completely filled with an epoxy resin or potting compound to provide the seal.

3.4 Ventilation

The quantity of air circulated to the various areas of the reactor building is based on (1) removing heat given off by equipment at a rate which will allow efficient and safe operation of the equipment, and (2) a sufficient number of air changes for the safety and comfort of personnel in occupied areas of the plant.

The air for the reactor building ventilation is taken from the outside. A fan and duct system distributes this air throughout the reactor building.

The air which is supplied to the operating and equipment floors is circulated around these areas and then exhausted to the atmosphere via the stack.

The air which is supplied to the plug floor circulates into each recirculation pump vault and into the sump pump vault through ducts penetrating the floor. A part of this air then circulates through pipe channels in the major shield wall into the bottom of the reactor vault which is maintained at a negative pressure when the ventilating systems are operating. A fan exhausts air from the top of the reactor vault through ducts which penetrate the major shield wall and connect to a main exhaust duct leading to the fan. The excess air in the pump and valve vaults which does not circulate into the reactor vault is drawn into the main exhaust duct by the same exhaust fan. This fan then discharges the air to atmosphere through a stack after it has been monitored for radiation level.

Supplementary ventilation is provided for the shield pool when refueling or maintenance operations are being performed. This ventilation system

consists of a fan and distribution duct which directs air over the pool surface to an exhaust duct. A fan connected to the exhaust duct discharges the air to the atmosphere through the stack.

To prevent the escape of contaminated air and steam in the event of an accident within the reactor building, an emergency 36 inch butterfly shut-off valve is provided in each duct that penetrates the building wall.

These emergency or "isolation" valves are furnished with a neoprene seat, and a gas cylinder operator. A solenoid switch, operated by an activity monitor located in the building trips the operator to close the valve. The main supply duct valve and the stack exhaust duct valve will close in about 4 seconds or less. The valves are designed to fail safe and upon power failure will close.

3.5 Testing

The primary purpose of the inspection and testing program is to provide assurance that the building will contain any radioactive substances which may be released by a rupture of the internal equipment. The secondary purposes of this program are as follows: to check the building for conformance with the plans and specifications; and to fulfill the requirements of the codes, referred to in Section 3.1, thus permitting this building to be stamped with the ASME code stamp. To accomplish the above purposes sufficient inspection and tests under close supervision by the contractor and the architect engineer and by a competent insurance inspection agency will be performed to meet the applicable codes. Records of all mill test reports for all steel used in the reactor building shell will be maintained. Specifications require the building contractor to maintain the markings

on plates identifying the heats from which the plates are made. Should more than one piece be cut from any plate, the markings will be transferred in presence of an authorized inspector to each piece prior to cutting. The tests as specified to meet current codes are as follows:

3.5.1 Pressure Tests

A pneumatic pressure test, at a pressure of 1-1/4 times the design pressure (97 1/2 psig), will be made as soon as erection of the shell has been completed. The purpose of this test, which is required by the ASME Code, is to insure the structural integrity of the building. After all penetrations have been completed, including all seals, expansion joints and the piping to which they are connected, and after the construction openings have been closed and welded, the second pressure test will be made. This test will be made at the same pressure as the first test. The purpose of this test is to insure the structural integrity of the seals and expansion joints and of the new welding.

3.5.2 Leak Rate Tests

Two leak rate tests will be performed in conjunction with the two pressure tests described above. The purpose of these tests, which will be conducted at design pressure (75 psig), is to demonstrate that the leakage from this vessel and its penetrations will not exceed 0.2% of the internal volume per 24 hours. Compliance with this construction specification will be demonstrated by maintaining the building at design pressure for 2 days with less than 0.4% loss of pressure as measured by a temperature compensated pressure measurement system similar to that used for other reactor building leak tests.

The first test will establish the leak rate for the building. If this test indicates a leak rate of over 0.2% of building volume per 24 hours, then the building contractor will examine the weld joints by the soap bubble method, the tracer gas method, or any other appropriate method until the leak or leaks are found and corrected.

The second test will establish the leak rate for the penetrations and seals. This test will be run simultaneously with the soap bubble check outlined later in this report.

The leak rate will be checked by use of a water manometer connected to closed drums suspended inside the vessel or an equivalent method. The method to be used will be submitted to the Inspection Division of the U. S. Atomic Energy Commission.

3.5.3 Soap Bubble Check

All joints where expansion bellows connect to pipes or to the building, and all seals will be checked by the soap bubble method. This check will be made simultaneously with the second leak rate test. To conduct this test, the outside (low pressure side) of each joint will be painted with a soap solution and the joint will then be observed for bubbles indicating leakage. All leaks will be repaired.

3.5.4 Inspection of Welds

Longitudinal and circumferential welds will be radiographed in accordance with the ASME Pressure Vessel Code as modified in specific rulings.

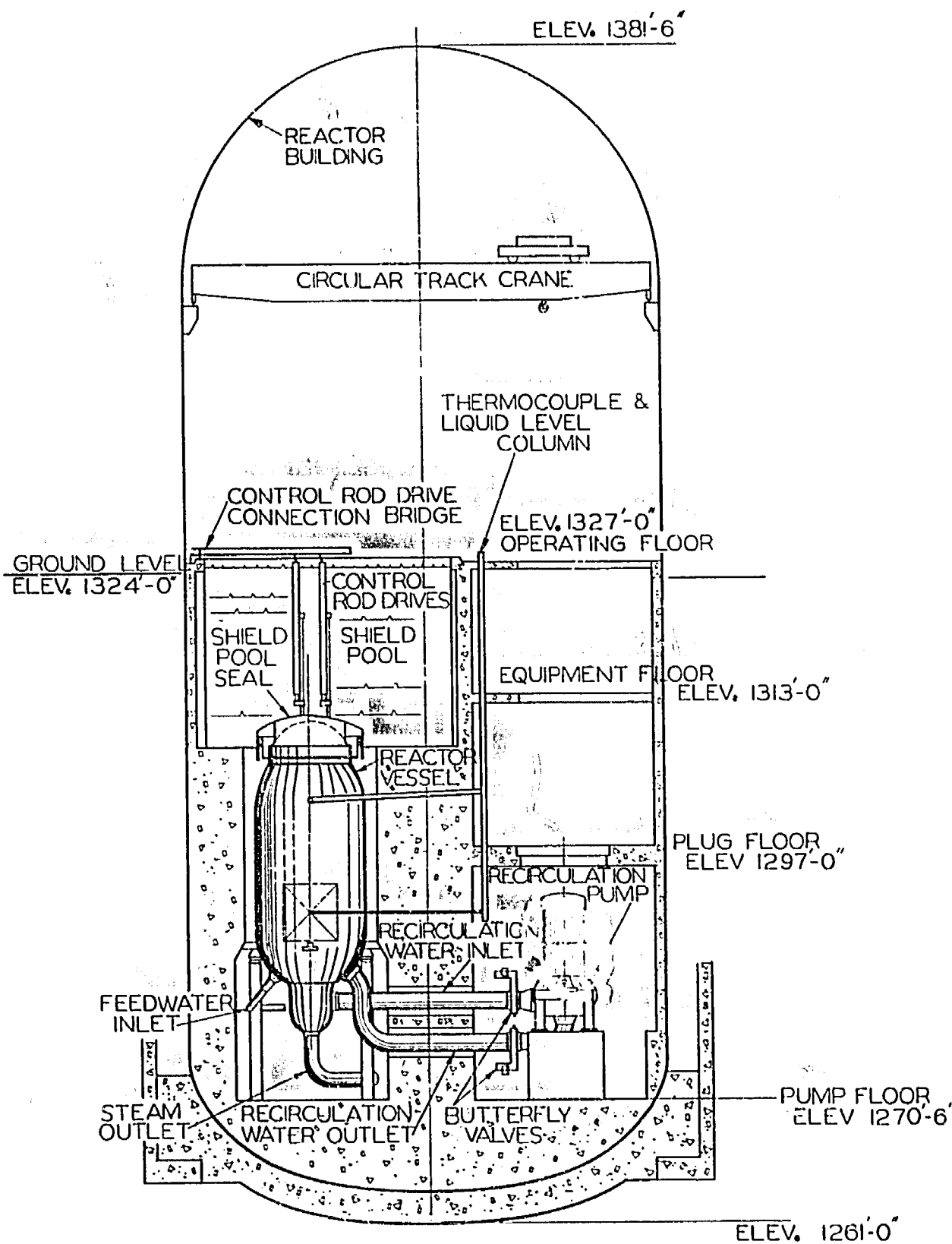
3.6 Missiles

The location of all equipment is such that were missiles to form they would be stopped by the structural and shielding concrete in the lower part

of the building before reaching the building wall. The only other conceivable "accident" that could produce missiles is the failure of high speed rotating machinery. No machinery of this type is housed in the reactor building. The recirculation pumps operate at relatively slow speeds. Machinery of this nature e.g., the turbine generator outside the reactor building, is oriented so that its plane of rotation does not intersect any part of the cylindrical reactor building. This orientation reduces the risk of damage to the reactor building to an acceptable level without special missile protection.

5.7 Thermal Shock

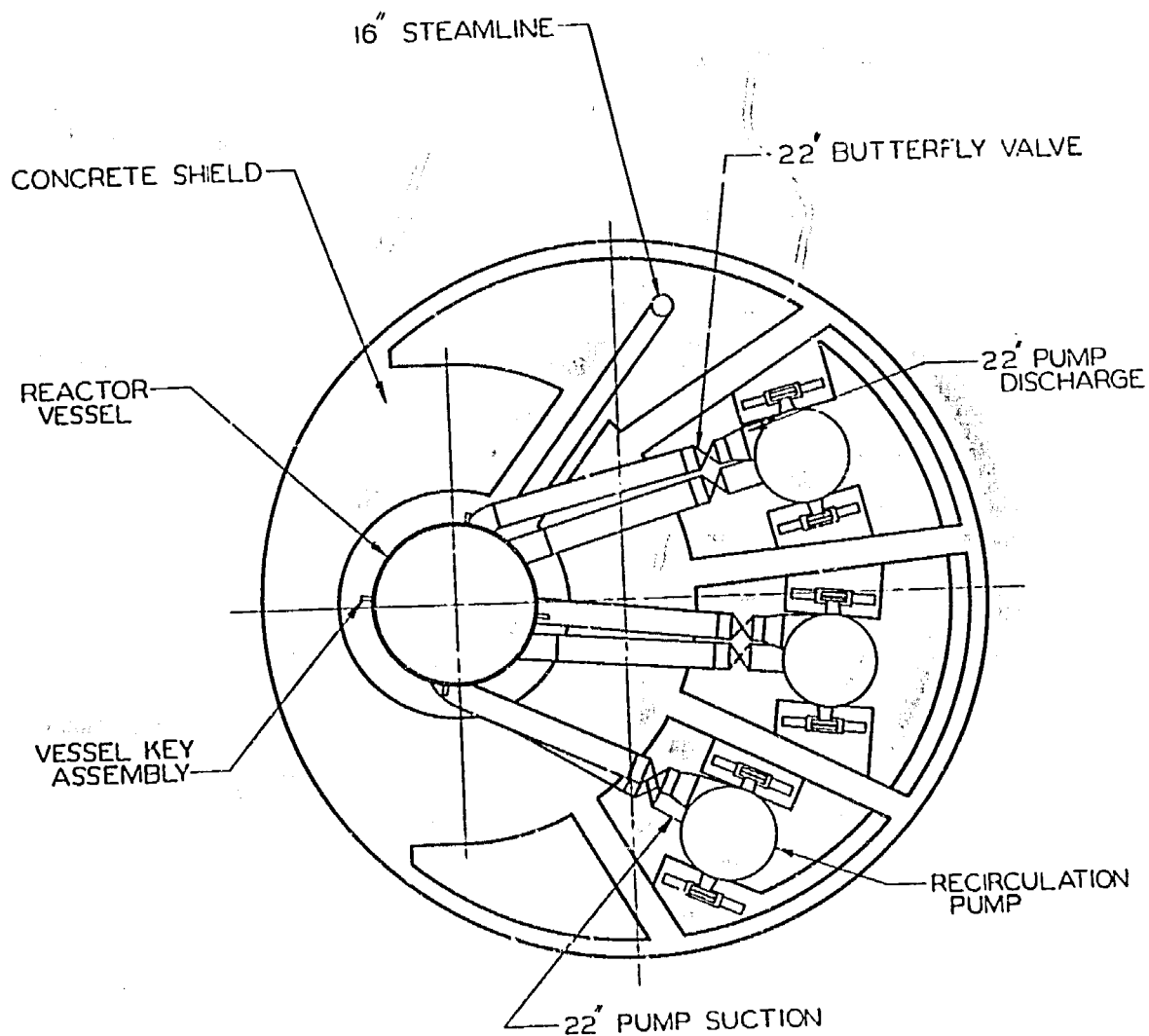
A failure in the primary reactor system would cause the temperature within the reactor building to rise reasonably fast. Certain areas of the building shell are restrained from free thermal expansion. This is especially true of that portion of the shell at, and, below grade level, however, these areas of the shell are not exposed to the high temperature. A pressure of 78 psig and the temperature existing under that condition does not represent a departure from existing experience. No unusual problems are anticipated. Specifications for the reactor building include the requirement that the contractor furnish computations in accordance with the provisions of the code to determine the thickness of the various parts, the reinforcement around penetrations and openings, and all details of design. These computations will be checked by the architect engineer and Allis-Chalmers.



REACTOR BUILDING

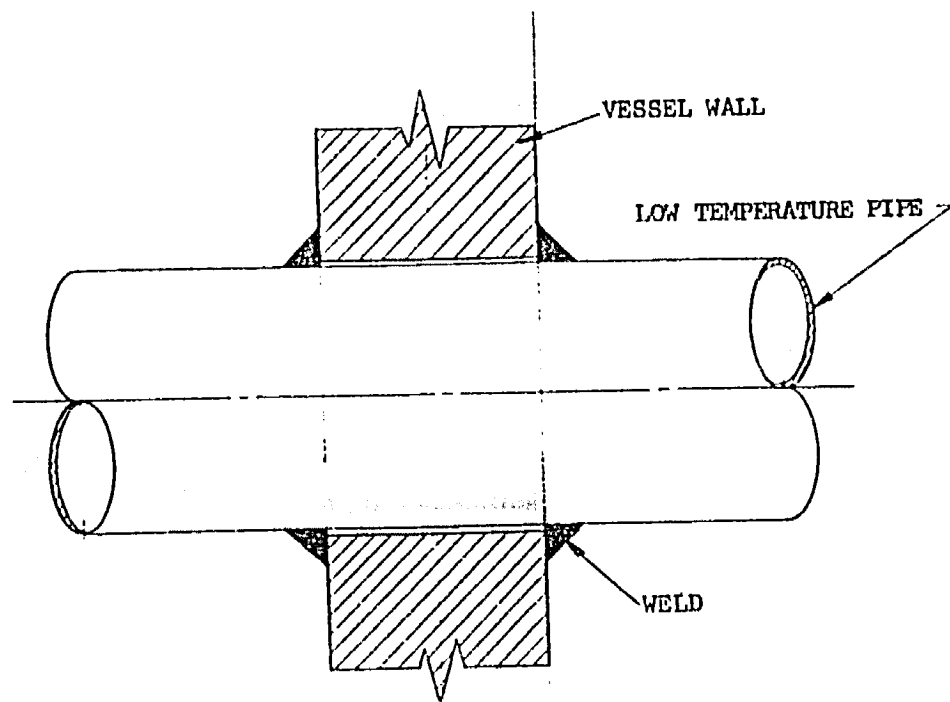
FIG. 3.1

August 15, 1960



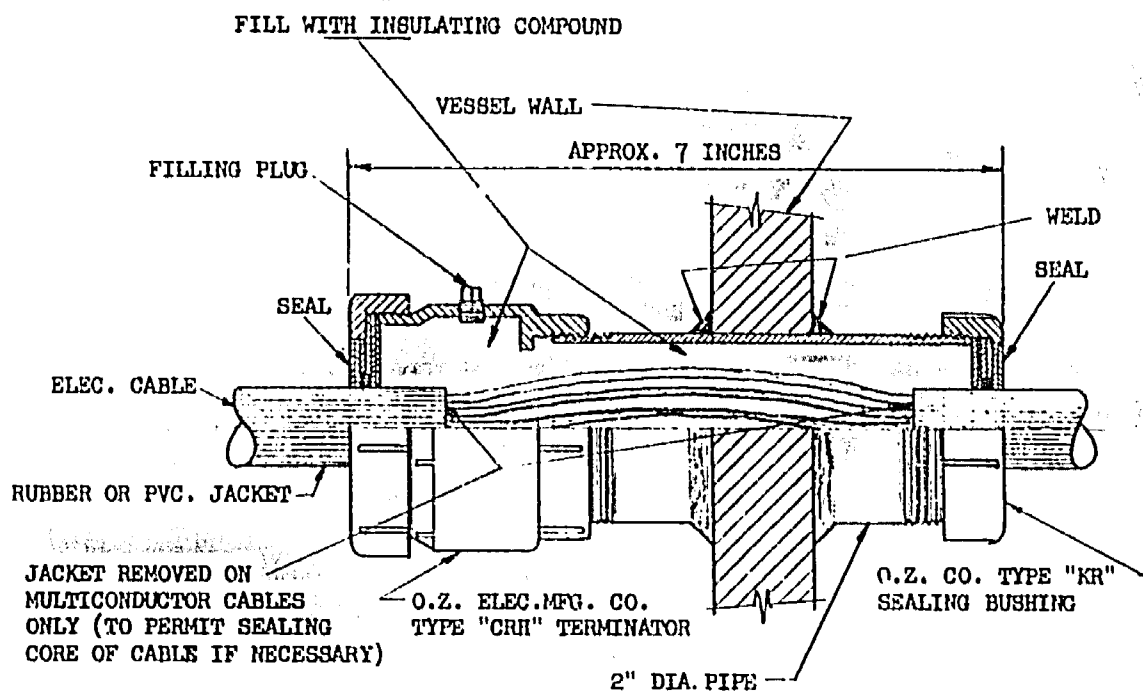
REACTOR WITH PUMPS IN REACTOR BUILDING

FIG. 3.2
August 15, 1960



RIGID SEAL FOR LOW TEMPERATURE PIPE

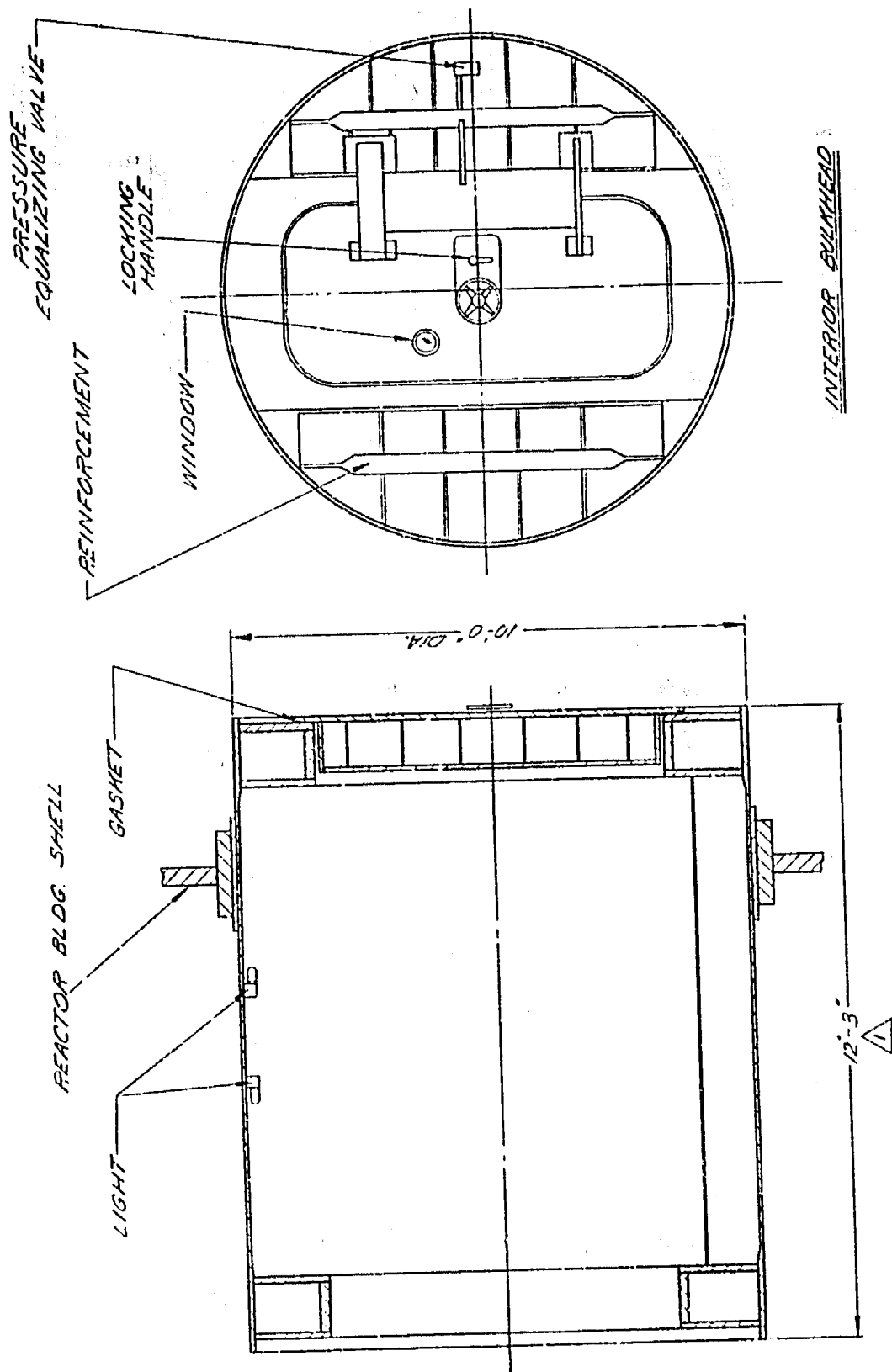
FIG. 3.4
August 15, 1960



SEAL FOR ELECTRICAL CABLE TYPE A

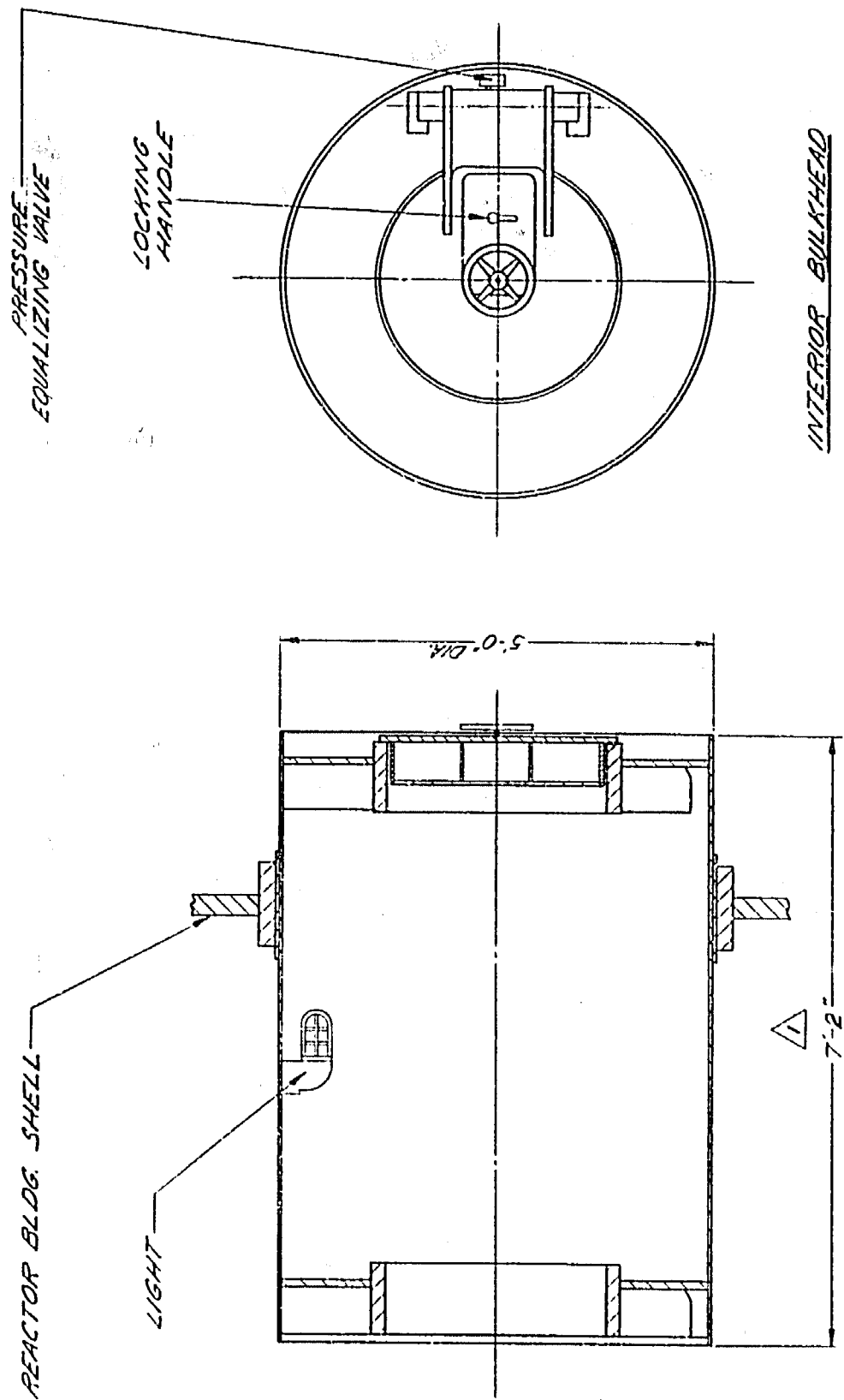
FIG. 3.5

August 15, 1960



PERSONNEL AIRLOCK

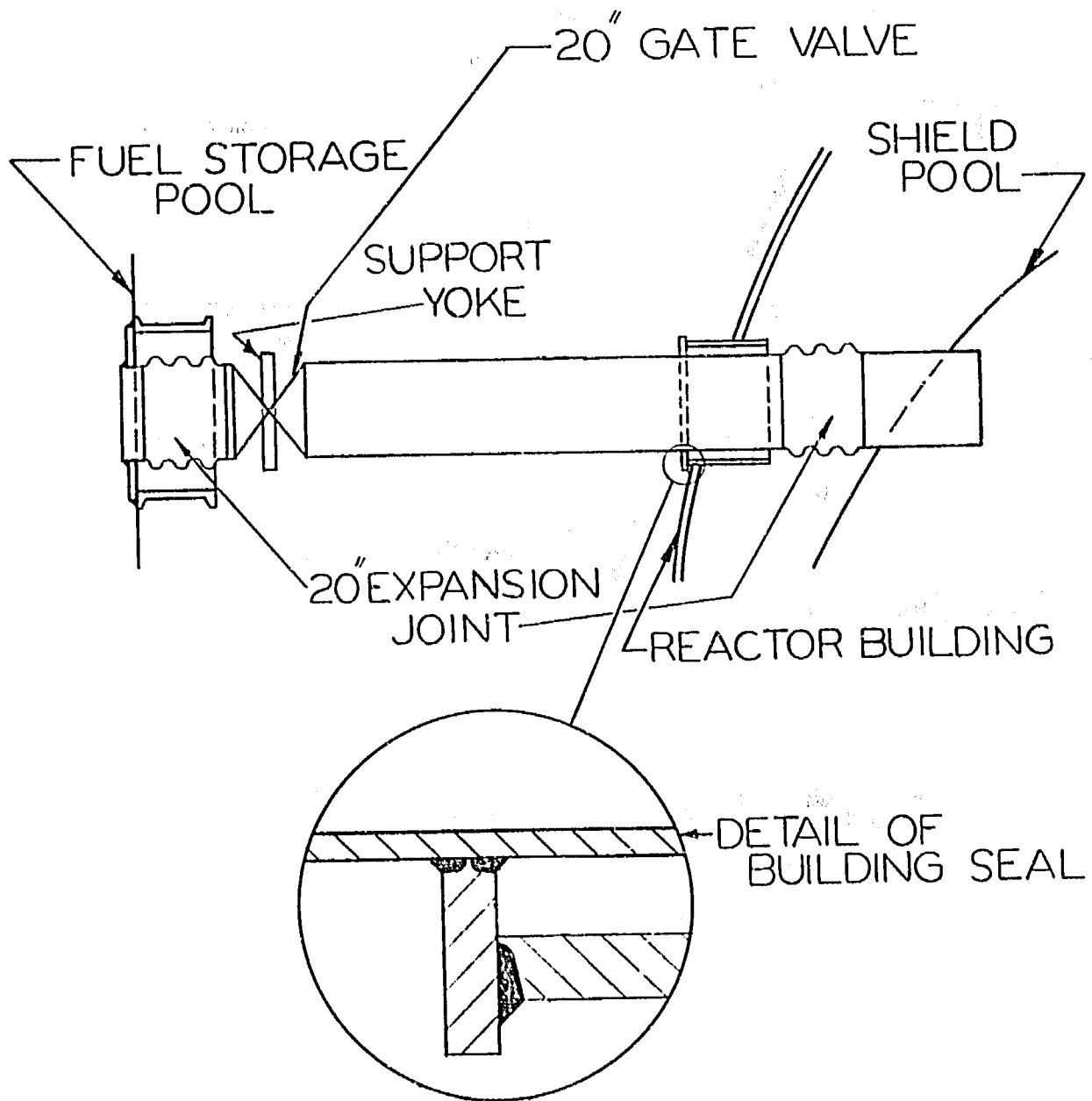
FIG. 3.6
August 15, 1960



EMERGENCY AIRLOCK

FIG. 3.7

August 15, 1960



PATHFINDER FUEL TRANSFER TUBE

FIG. 3.8
August 15, 1960

- 7) Recirculation pump and butterfly valve tests to insure proper operation of those units and their controls.
- 8) Tests of all other devices and auxiliaries associated with the primary system to assure their proper functioning. Included in this series of tests are those of the liquid level indicators, and the valves in the various lines connected to the primary system.

At the end of these tests, all equipment associated with the operation of the reactor and its safety system will have been checked and its performance demonstrated. Any piece of equipment that does not meet its specifications will be adjusted, repaired, or replaced as necessary to insure safe operation of the plant.

6.7 Initial Criticality and Experiments

With the instrumentation and controls in proper working condition, the reactor will be ready for loading of fuel. Operating personnel will be guided by detailed operating procedures, a digest of which is contained in the following subsections. In addition to the reactor instrumentation necessary for the normal operation of the plant, the initial Pathfinder core loading will include in-core instrumentation in the superheater with suitable monitoring or recording equipment to be used for initial start-up and oscillator tests to determine the core operating conditions and parameters insofar as is practical. Approximately nine in-core ion chambers as well as the normal chambers outside the core will be used to closely monitor the core activity. About forty special fuel element assemblies with thermocouples attached are included in the first superheater loading. Information obtained from this instrumentation will be available to the operator during the testing period. The operator will utilize

this information and limit plant operation so that predetermined maximum conditions such as temperatures, pressure or flux are not exceeded.

Critical facility experiments have already been performed on an equivalent core at Allis-Chalmers Greendale Laboratories and the personnel supervising the loading at Padriindar will be familiar with the results of these experiments.

6.7.1 Initial Loading

The general procedure for the initial loading is as follows:

- 1) All the control rods which will be effective in the control of the core region being loaded will be placed into their proper position in the core.
- 2) The in-core ion chambers will be placed in superheater fuel element channels and the leads secured. (No fuel will be added to these channels.) Instrumentation will include special removable start-up chambers located within the core during the loading operation.
- 3) The forty special instrumented superheater fuel elements (positively known to be a subcritical loading) will be placed in the core and the thermocouple leads secured. (The reason for loading these elements prior to the neutron source is so that the leads can be properly tied down).
- 4) The reactor vessel will be filled with water to a level above the top of the core.

- 5) The neutron source will be placed in the core and flux levels will be established on the monitoring instrumentation.
- 6) Fuel will be loaded into the core in increments, starting with the superheater core region in the center and the loading built outward through the boiler region. The loading will be kept cylindrical insofar as is practical as the loading proceeds.
- 7) After each increment of fuel is loaded, the vessel cover will be set on the vessel, approximately four of the nuts will be turned finger tight on the flange studs and the appropriate control rod drives will be latched to the control rods in the loaded core region.
- 8) The control rods in the fueled region will be raised as a bank. The height of all rods will be kept as uniform as practical. The activity of the core will be monitored and its reciprocal plotted against fuel loading so that the approach to criticality can be predicted in the conventional manner.
- 9) The bank height required to reach criticality (if it is attained) will be plotted against fuel loading in the core after each increment of fuel is added. As the loading progresses, the control rod bank height required to reach criticality will decrease to about fifteen inches (measured from the bottom of the core) at full loading.

The first Pathfinder core will include four special boiler fuel element assemblies and a drum type rotating oscillator rod in place of one of the normal cruciform control rods.

- (10) After all the fuel has been loaded into the core, the vessel cover will be fastened to the vessel in the proper manner and all the instrumentation lead penetrations sealed.

6.7.2 First Core Experiments

Phase I tests (1 KW thermal or less). A series of essentially zero power experiments will be performed to confirm the results of the critical experiments previously performed at the Allis-Chalmers Greendale Laboratories. These will include control rod worth tests, flux distribution measurements, temperature and superheater flooding coefficient tests, and a zero-power transfer function experiment. No coolant flow will be required for the reactor. The recirculation pumps will be used on occasion, however, to help heat the system as required for some of the experiments.

6.8 Initial Start-up

Phase II tests (5 Mw thermal or less). As part of the pre-construction research and development program, a number of heat transfer tests were performed (See Sec. 1.1.2) to determine the ability of the nuclear superheater fuel to cool itself by radiation heat transfer to the moderator in the event of loss of coolant. The results of these earlier tests will be checked in the reactor. The superheater will be drained and the steam line isolation valves closed so that no steam flow can occur. The reactor power will be raised in increments. At

each increment, the superheater hot spot fuel temperature and moderator temperature will be carefully monitored and recorded. During these tests the hot spot temperature of the superheater fuel assemblies will be monitored by in-core thermocouples and limited to about 1400 F. Reactor pressure will be limited to about 550 psig.

Curves of both the superheater fuel temperature and moderator temperature will be plotted versus reactor power as the test progresses. The test can be readily terminated by scrambling the reactor and opening the steam line isolation valve if the need arises.

Other tests included in this series will be void, temperature and flooding coefficients and low power transfer function experiments at various operating pressures to demonstrate the stability of the reactor. At this time, radiation levels throughout the plant and activities in the off-gas system will be checked. The performance of the emergency heat exchanger will also be checked by scrambling the reactor from about 5 MW and intentionally isolating the reactor system.

Phase III (Full power or less)

These tests will be a repeat of the transfer function experiments at increasingly higher powers. For convenience, steam flow may be bypassed to the condenser. The superheater-boiler power ratio and transient superheater hot spot temperature will be carefully monitored. During this period, the design calculations will be checked against experimental results. At the completion of these experiments, the oscillator rod will be removed from the reactor and replaced by a regular rod and drive.

Phase IV (Full power or less)

The final series of tests will extend over several months and will consist primarily of observations of the reactor and system under steady state conditions at various powers. A final check will be made of the radiation levels throughout the plant and an evaluation of both the gaseous and liquid waste handling systems. After the equivalent of about three months of full power operation, or sooner if steam line activity indicates a fuel element failure, the reactor will be shut down and a number of fuel elements removed from the reactor and visually inspected. The exact number will depend on the observed condition. If the inspected superheater and boiler elements show no abnormal effects from operation, they will be returned to the reactor and run for the equivalent of another three full-power months or until a fuel failure occurs. If the fuel assemblies removed from the reactor show evidence of excess corrosion or erosion, blistering, or other indication of potential failure, a larger sampling of them will be removed from the reactor. If the condition of any of the elements is judged to be poor, at least one such element will be destructively examined in a thorough manner. If the general condition of the elements is considered to be poor the fuel will be replaced.

After a total of six full-power months, the inspection procedure will be repeated. As discussed in subsection 1.1.2 of this report, no superheater fuel failures are expected before six months. At this time at least one superheater element assembly selected at random from the hot channels will be destructively inspected to evaluate the adequacy of

the design in regard to such items as cladding, tolerances, clearances, bonding, and fuel alignment. This inspection will provide the information needed to make a reasonable prediction of the remaining element life. In the meantime, the removed element or elements will be replaced with new fuel and reactor operation resumed.

The initial Pathfinder fuel loading will be replaced after about nine full power months for the superheater and about 10,000 MWD/T for the boiler unless there is evidence that it must be replaced sooner. The second Pathfinder fuel loading is expected to be identical to the first except that all the in-core instrumentation will be omitted.

Phase IV tests will also include irradiation of structural material samples in the reactor, such as samples of materials used for superheat grid structure, superheater manifold, and reactor vessel, and periodically examine them for embrittlement.

6.9 Normal Startup

It is assumed that the conditions before startup will be similar to those following a refueling. They are:

1. Reactor vessel open to shield pool.
2. Core refueled, viz, one third new.
3. Reactor and shield pool water at about 105F.
4. Reactor shut down approximately 30 hours or longer.
5. Decay heat rate approximately 2.7×10^6 Btu/hr.
6. Decay heat being removed by water flowing by gravity from the reactor shield pool to the main condenser and returned to the reactor by the condensate pump. (Figure 2.1)
7. Condenser operating at atmospheric pressure.

which will open if the operator releases it. Should this circuit fail, the drive would continue to pull a control rod out of the reactor. The operator can control the situation by turning the selector switch to another rod drive thereby interrupting the original circuit and allowing him (or the automatic control system) to insert a compensating rod. If he fails to do this the rod will continue on out and insert a maximum of $1.5\% \Delta k/k$ reactivity (the maximum worth of a rod).

If this incident occurs while the reactor is in the power range, the reactor power will rise. Additional steam voids will form and a higher fuel temperature will occur, both of which subtract reactivity and thus limit the power rise. The incident will be terminated by a high power level scram if not caught sooner by the operator. If the pressure reaches 660 psig, the pressure relief valves will open.

If the incident occurs while the reactor power is near the shutdown level, then the reactor can be placed on a relatively short period. Here the period signal will scram the reactor if the operator fails to do so. Should all equipment fail, the incident could result in an excursion. It is planned to analyze this excursion as the plant design program progresses.

7.3.7 Fuel Element Failures

The consequences of fuel element failures in the boiler region of the core are orders of magnitude less severe than those occurring from failures of fuel elements in the superheater region because the boiler water will retain a significant portion of the released radioactivity. The nature of the superheater also makes it more prone to fuel failure than the boiler because it depends on the boiler for its steam coolant. Its coolant flow is also more readily affected by influences from outside the reactor such as equipment failures. Therefore, the following discussion is limited to superheater fuel failures with the understanding that boiler failures would not be as great a problem.

Normal Leaks

As discussed in Section 1, no catastrophic failures have thus far been experienced nor are any expected with the cermet type superheater fuel. There may be a progressive swelling of the fuel due to irradiation damage which in some cases might lead to cracks or small blisters with cracks in them if the fuel is burned too long in the reactor. These failures are expected to occur in a random manner and experience to date indicates that the resulting blisters cannot bridge the steam flow passages and lead to catastrophic melt-down or further fuel failures.

As fuel burn-up proceeds and fission products build up, the gases and volatiles exert an internal pressure that eventually may cause this cracking or blistering. When a crack occurs, these gases and volatiles will be released to the steam coolant and cause a radioactivity increase in the steam line which can be detected by the plant instrumentation long before dangerous levels are reached. The random failure of superheater fuel element near the end of life,

If they occur, will cause a gradual radioactivity increase in the steam above the N-16 background. Before the activity approaches dangerous levels or for other considerations, the operator will normally shut the plant down. If he should neglect to do this, the reactor will be scrammed automatically when dangerous levels are reached and the reactor isolated.

During operation of the Pathfinder reactor the superheater fuel will be subjected to variations in steam flow. These will be discussed in Section V of this report. In general, these fluctuations are of very short duration (less than a minute). During the early months of operation, short term increases in the hot spot fuel temperature as a consequence of momentary reductions in steam flow are not expected to cause damage to the fuel. However, near the end of life after fission gas pressures have built up, it is possible that a momentary increase in temperature of a hundred degrees or so at the hot spot may cause the fuel to fail if it is already near the failure point. These failures would be no worse than the normal leaks already discussed except that a number of these failures may occur simultaneously. If excess activity is detected in the steam line, the reactor will be scrammed automatically.

The released fission products carried by the steam, will otherwise be transported to other parts of the power system. The bulk of the volatiles such as bromine and iodine (and solids if any) are expected to plate out on the available surfaces of the system or combine with the water in the condenser. Some may be recycled into the reactor but are more likely to be trapped by the feedwater filter. Some of those returning to the reactor will be taken out by the purification system.

The non-condensable fission products would be carried out through the air ejector into the gaseous waste disposal system which is designed to accommodate a certain level of them. As discussed in Section 2, gas decay storage tanks and the length of piping leading to the stack provide from several minutes up to 18 hours delay depending on the degree of gas throttling at the discharge to the stack and the system pressure. (24 hours delay is actually provided, however, here 5 hours are considered for a normal plant shutdown). If the activity level becomes too high for discharge to the atmosphere after the gases pass through the gas disposal system, the discharge valve will automatically be closed. Again, assuming a gradual increase in fission product release, the operator will have adequate warning and can take appropriate action before automatic shutdown occurs. The plant has been designed so that a total of 500 Mg of U-235 can be exposed without exceeding the maximum permissible concentration of airborne radioactive material at the site boundary. The cermet structure is impermeable to fission products so only that fuel beneath a blister or cracked area need be considered as exposed. Assuming an exposure of fuel in an area 1/8 inch in diameter (as indicated by irradiation experiments) the steady state release from 100 such ruptured areas can be tolerated at full power operation without exceeding allowable discharge limits.

Large Crack or Rupture

Although no evidence exists to warrant anticipation of a sizable fracture wherein a burst of fission products is released from the fuel, such an accident is postulated there for analysis. The burst of fission products passing the steam line radiation monitor will cause it to initiate a reactor scram. If the activity in the line persists at a high level, the steam line isolation

valves will close and the reactor placed on the emergency coolant system within the reactor building. Some high activity in the meantime, will have progressed to the condenser. Again, particulates and volatiles would be expected to be trapped there. The gases would be vented to the gas disposal system.

A line monitor is located on the pipe exiting from the air ejector. At the location of the monitor the pipe has provided three minutes of decay time for the off-gas. The monitor, on reading a high radiation level in the off-gas will cause closure of the valve located in the line which connects the gas disposal system to the stack. This valve can be closed in about one second while the transit time of the gas through the system to the valves is a matter of minutes or longer. Thus, the system is isolated before any high activity is released to the atmosphere. A second monitor at the stack provides another safety feature in that it will close the discharge valve and limit the discharge should the first monitor fail to function. It can be seen that suitable provisions for containment of released fission products are provided; first, by means of the reactor building isolation valves which close and contain the activity within the reactor building and, second, by means of the off-gas system isolation valves which close and prevent the atmospheric release of any gases escaping the primary system. Therefore, no hazard to operating personnel or to the public is expected from the type of accident discussed above.

Catastrophic Fuel Failure

As discussed in other parts of this report, no fuel meltdown is expected from other than the postulated maximum design accident. To obtain an indication of the consequences of superheater fuel meltdown, it is postulated here that a single superheater assembly, at equilibrium burn-up, melts completely, and

further that the fuel and fission products are carried by the steam out of the reactor building through the steam line. If all of the fission products are deposited in the low pressure turbine, the dose rate 3 feet from the outside surface would be 330 r/hr at the maximum dose point. Similarly, if all the fission products were uniformly distributed throughout the condenser, the dose 3 feet from the surface would be 10 Y/hr. A more likely distribution, however, would locate the solids, condensibles, and trapped gases in the condenser hot-well and the residual gases in the off-gas system. If all the gases were to escape the fuel (which is not likely) and stop in the air ejector and if the particulates were to stay in the hot well, the dose rate 3 feet from each of these pieces of equipment would be 23 and 20 Y/hr respectively. Since actual distribution of fission products throughout the system is not easy to predict, the above dose rates can be used as conservative upper limits. High activity would be contained in the primary and the gas disposal systems because of the provisions previously discussed and thus can be adequately dealt with without endangering the operating personnel or the general public.

Failure or malfunctioning of equipment external to the reactor may indirectly affect fuel element integrity. These incidents are discussed in other subsections.

7.4 Structural Failure of the Primary System - Hypothetical Incidents.

7.4.1 General Discussion

The reactor vessel and piping will be designed and tested according to the appropriate codes. This will include radiographic and dye penetrant inspection and hydrostatic pressure tests.

The system will be equipped with pressure relieving devices so that no

overstressing of the material will take place. The operating conditions are essentially steady thus no large pressure and temperature fluctuations are expected. Piping will be so supported that excessive stresses are not imposed on them or their connections. The recirculation loops will be suitably suspended with adequate freedom of movement. The design will minimize vibration in the recirculation loops and minimize expansion forces and moments. Pipes penetrating the reactor building wall will be looped or have expansion joints to provide for thermal expansion.

Types of failure initiated by the structural conditions in the pressure containing material were studied for the conditions existing in the Pathfinder Plant. An attempt was made to visualize all conceivable circumstances which could cause localized loads in excess of the allowable limits. The probability of their occurrence is evaluated and preventative design features are discussed.

Creep - Allowance is made for creep in determining allowable design stresses. Failure from creep is preceded by a period of time during which gradual deformation takes place.

It is believed that failure through creep is not a feasible accident in the CRBR system. This conclusion is based upon the following considerations:

- 1) The temperature of pipe and vessel walls is expected to be low, not exceeding 550 F in most areas. Creep is unimportant for steels at these temperatures. Main and auxiliary steam lines, and the emergency heat exchanger system are exposed to super-

SOUTH DAKOTA
Department of
Water & Natural Resources
Jon Foss Building
Pierre, South Dakota 57501

1-3, 2-3, 3-3, 4-3, 7-3, 8-3, 41-3
42-3, 43-3, 61-3, 70-3, 519-3, 519-3A,
561-3 & 742-3

August 15, 1979

Neal Nelson-Construction Manager
Northern States Power Co.
Box 988, 500 W Russell St.
Sioux Falls, SD 57101

Dear Mr. Nelson:

This will confirm our discussion regarding water rights presently held by Northern States Power Co. in the Sioux Falls area.

Permit No. 1-3 was issued with a diversion of 500 cfs at a point in the NE $\frac{1}{4}$ SW $\frac{1}{4}$, Section 16, T101N, R49W, with a priority date of July 2, 1907.

Permit No. 2-3 was issued with a diversion of 500 cfs at a point in the NE $\frac{1}{4}$ SW $\frac{1}{4}$, Section 16, T110N, R49W, with a priority date of July 2, 1907.

Permit No. 3-3 was issued with a diversion rate of 400 cfs at a point in the NE $\frac{1}{4}$ SW $\frac{1}{4}$, Section 16, T101N, R49W, with a priority date of July 2, 1907.

Permit No. 4-3 was an uncompleted permit to use up to 300 cfs from the Sioux River in Hamlin County at a point in the NE $\frac{1}{4}$ SW $\frac{1}{4}$, Section 19, T114N, R51W.

Permit No. 7-3 was issued with a diversion rate of 500 cfs at a point in the NE $\frac{1}{4}$, Section 16, T101N, R49W, with a priority date of September 17, 1907.

Permit No. 8-3 was issued with a diversion rate of 500 cfs at a point in the NE $\frac{1}{4}$, Section 16, T101N, R49W, with a priority date of September 17, 1907.

On August 21, 1912, the Sioux Falls Power and Light Company petitioned the State Engineer to consolidate permits 1-3, 2-3, 3-3, 7-3 and 8-3. This action was approved by the State Engineer on August 24, 1912.

With the Abandonment of the Hydro Operations, by resolution dated December 6, 1961, Northern States Power Co. surrendered and abandoned its right to appropriate 2,325 cfs of the 2,400 cfs authorized under Permits No. 1-3, 2-3, 3-3, 7-3 and 8-3, retaining 75 cfs for operation of the steam facilities.

A letter from G.V. Velk dated August 9, 1974 indicated Northern States Power Co. had conveyed its Sioux Falls Generating Plant and associated land to the City of Sioux Falls, and requested that the 75 cfs retained under the above permits also be transferred to the City. Inasmuch as the water rights held by the Northern States Power were essentially non-consumptive, it was determined that they would be of no value to the City, and the Permits could be cancelled.

Equal Opportunity Employer

Under the above relinquishment and abandonment, the Permits 1-3, 2-3, 3-3, 7-3 & 8-3 all pertaining to the Hydro Plant and subsequently the steam plant, had not been officially cancelled by the Water Rights Commission. In official action by the Commission the above permits were cancelled on November 14, 1974. Uncompleted Permit No. 4-3 was also cancelled under this action. The above permits except for 4-3 were used in connection with the Bennett Dam and/or what you referred to as the Weber Plant.

The following Water Rights were issued in connection with the Lawrence Plant.

Water License No. 41-3 covers Well #1 and authorizes diversion of .11 cfs (500 gpm) with a priority date of July 18, 1946.

Water License No. 42-3 covers Well No. 2 and authorizes diversion of .067 cfs (30 gpm) with a priority date of July 18, 1946.

Water License No. 43-3 covers diversion from the Sioux River authorizing diversion of 2.67 cfs (1200 gpm) with a priority date of January 17, 1947.

Water License No. 61-1 covers Well No. 3 and authorizes diversion of 1.0 cfs (450 gpm) with a priority date of October 10, 1960.

Water License No. 70-3 covers Well No. 4 and authorizes diversion of 1.55 cfs (700 gpm) with a priority date of April 21, 1951.

In our discussion you indicated that the Lawrence Plant had been deactivated on September 30, 1977, but because of energy requirements, that it may be reactivated. South Dakota Water Laws state that a 3 year period of non-use is a basis for forfeiture of water rights. If the plant is reactivated prior to September 30, 1980, the water rights would be valid, and remain in force as long as the water is used beneficially. We request that you notify us if the plant is reactivated.

The following water rights were issued in connection with the Pathfinder Plant.

Water License No. 519-3 covers diversion from the Sioux River, and authorizes diversion of 6.0 cfs, with a priority date of February 10, 1958.

Water License No. 519-3A covers diversion from the Sioux River (enlargement of 519-3) and authorizes diversion of 1.77 cfs (700 gpm) with a priority date of December 12, 1962.

Water License No. 561-3 covers Well A, and authorizes diversion of 1.77 cfs (700 gpm) with a priority date of December 18, 1958.

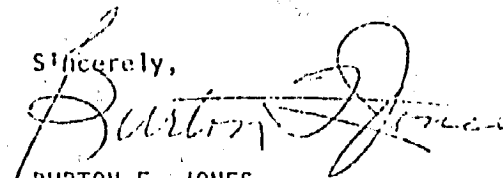
Water License No. 742-3 covers Well B and authorizes diversion of .33 cfs (150 gpm) with a priority date of September 17, 1960.

These permits were primarily for cooling water and plant use for the atomic generators. You indicated that the plant has now been converted to an oil burning plant, and the water is used in connection with this operation. These are considered valid permits and remain in force as long as the water is used beneficially.

page 3.
August 15, 1979

This summarizes the water rights that are presently active and disposition of those no longer used. We would like to thank you for the time spent discussing the status of your existing use. If you have questions, feel free to contact this office.

Sincerely,

A handwritten signature in dark ink, appearing to read "Burton F. Jones", written over a horizontal line.

BURTON F. JONES
Water Rights Specialist

BFJ/gy



519-3
DEPARTMENT of ENVIRONMENT
and NATURAL RESOURCES

PMB 2020
JOE FOSS BUILDING
523 EAST CAPITOL
PIERRE, SOUTH DAKOTA 57501-3182
denr.sd.gov

August 6, 2012

Dear Water Right Holder;

Due to low flows I am advising water right holders on the Big Sioux River of the possibility of shutoff orders in 2012 for the river reach from Watertown through Sioux Falls. This is **not a shutoff order**.

As a result of the continuing lack of precipitation, tributary flows throughout the Big Sioux River basin remain well below the historical average. The forecast for improved conditions for the near future is not favorable.

This letter is only intended to give you advanced warning that your ability to divert water may be cut short this year. If shut-off orders are issued, you will receive another letter from the department. In the meantime, water right holders can help by closely monitoring water usage and trying to avoid wasting water.

South Dakota water law gives the highest priority to domestic use, which includes livestock watering. There is generally adequate water available to meet both domestic use and other needs. However, once water availability reaches critical low levels, other uses, including irrigation, will begin to interfere with domestic use. If levels become critically low, the department must issue shut-off orders to protect domestic water uses. Once these orders are issued, the department will continue to monitor water availability and should conditions improve, the shutoff orders would be rescinded.

If you are interested, the US Geological Survey has an Internet site showing real-time river flows at 159 stream gauging stations across the state including three gages sites on the Big Sioux River. To access this information go to <http://waterdata.usgs.gov/sd/nwis/rt> and click on the link "Statewide Streamflow Current Conditions Table".

Thank you for your cooperation. If you have any questions, please contact Lynn Beck or Mark Rath at (605) 773-3352.

Sincerely,

Garland Erbele, Chief Engineer
Water Rights Program

CERTIFICATION

I hereby certify that on August 6, 2012, I have personally deposited with the United States mail at Pierre, South Dakota, first class postage, prepaid envelope(s) containing a Notice dated August 6, 2012 regarding shut off warnings on Big Sioux River addressed as stated below:

permitno	fname	lname	addrss2	city	state	zip
1696-3	DONALD G	AHLERS	22920 SD HWY 13	FLANDREAU	SD	57028
5530-3	TERRY & DENISE	ANDERSON	47558 237TH ST 26534 N SHORE PLACE	TRENT	SD	57065
6192-3	BERNE B	BAHNSON	PO BOX 366	HARTFORD	SD	57033
3806-3	BERNARD	BERGDALE	47948 232ND ST	ALCESTER	SD	57001
1175-3	HAROLD	BOEVER	4162 BIRCH AVE	FLANDREAU	SD	57028
4895-3	BILL B	BOYER	403 16TH ST	HAWARDEN	IA	51023
284-3	LUKE	BOYER, MGR	24482 482ND AVE	HAWARDEN	IA	51023
7110-3	BRIAN	BURKHART	32650 480TH AVE	DELL RAPIDS	SD	57022
5760-3	BRIAN	CHICOINE	29509 488TH AVE	ELK POINT	SD	57025
2255-3	JAMES H	COLE	47364 GOLF VIEW DR	HUDSON	SD	57034
6124-3		DELLS ROCKY RUN GOLF COURSE		DELL RAPIDS	SD	57022
4969-3	JOHN GILMAN	FRAZIER FARMS	BOX 423	PARKSTON	SD	57366
4969-3	JOHN TROY D & SANDY F	GILMAN	4162 BIRCH	HAWARDEN	IA	51023
249-3	DARREN & ALLISON	HAHN	6762 SMITH DR	WENTWORTH	SD	57075
2065-3		HAMILTON	22615 480TH AVE	FLANDREAU	SD	57028
2952-3, 956-3	WAYNE	HEADRICK	47926 SD HWY 32	FLANDREAU	SD	57028
279-3	ROBERT	JAGER, MGR	48495 299th ST	HAWARDEN	IA	51023
1350-3, 6042-3	RAYMOND D	JENSEN	24348 478TH AVE	DELL RAPIDS	SD	57022
1302-3	LARRY J	JORGENSEN	24048 477TH AVE	TRENT	SD	57065
1500-3, 213-3	TED	LACEY	24064 478TH AVE	TRENT	SD	57065
	EUGENE	MOLLET	48011 332ND ST	JEFFERSON	SD	57038
3433-3	VIRGIL L	MOLLET JR	47991 334TH ST	JEFFERSON	SD	57038
1801-3	GAIL	MUNK	803 DAKOTA	DELL RAPIDS	SD	57022
113-3	PATTI	NASH	21660 469TH AVE	BROOKINGS	SD	57006
3914-3	OMAR F	PETTERSON	48378 276TH ST	CANTON	SD	57013
2350-3		RIVER VALLEY FARMS	PO BOX 486	ALCESTER	SD	57001
1801-3	HARVEY JOHN T & MELISSA R	SCHMIDT, OPERATOR	47528 244TH ST	DELL RAPIDS	SD	57022
1454-3		SCHUTTE	47973 W 5TH ST 47844 COUNTY RD 1B	CANTON	SD	57013
5814-3	PATRICK V	WALSH, OPERATOR	11726 NORWAY ST NW	ELK POINT COON RAPIDS	SD MN	57025 55448
597-3	DAN & JUDY	WHIPPLE WILLOW RUN MANAGEMENT COMPANY	8000 E HWY 42	SIOUX FALLS	SD	57110
5055-3		XCEL ENERGY	7100 E RICE ST	SIOUX FALLS	SD	57110
519-3A		NORTHERN STATES POWER COMPANY	414 NICOLLET MALL	MINNEAPOLIS	MN	55401
43-3, 519-3 5698-3, 6210-3	GREG ANDERSON	CITY SIOUX FALLS	224 W 9TH ST	SIOUX FALLS	SD	57102

Gail Jacobson

Gail Jacobson

Secretary/Water Rights

STATE OF SOUTH DAKOTA)

) SS

COUNTY OF HUGHES)

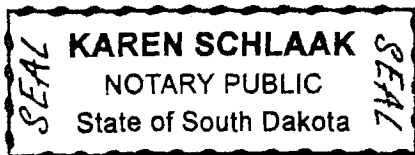
Sworn to, before me, this 4th day of August, 2012.

Karen Schlaak

Karen Schlaak

Notary Public

My Commission expires April 1, 2013





DEPARTMENT of ENVIRONMENT
and NATURAL RESOURCES

JOE FOSS BUILDING
PMB 2020
523 EAST CAPITOL
PIERRE SOUTH DAKOTA 57501-3182
www.denr.sd.gov

January 31, 2018

43-3 519-3 519-3A

NORTHERN STATES POWER COMPANY
414 NICOLLET MALL
MINNEAPOLIS MN 55401

Subject: Bypass Qualifications on Big Sioux River Water Rights

Dear SIR,

I am writing to inform you of an effort the SD Dept. of Environment and Natural Resources Water Rights Program is undertaking to review and update a number of water right permits authorized to pump water from the Big Sioux River. Since your water right does not have a bypass qualification, we **are not** proposing to amend your permit by adding a bypass qualification. However, I do want to make you aware of what we are proposing for several other junior priority Big Sioux River permits.

When a number of these Big Sioux River permits were issued, they contained a qualification requiring a minimum flow bypass before the user is able to pump water from the river. The bypass qualification was placed on these permits to protect the use of water by downstream senior priority water right holders and domestic water users, which includes livestock watering. The review found, with the passage of time, a number of senior water right permits have been cancelled due to abandonment or forfeiture. Consequently, it may be possible to lower the flow bypass limitation on the remaining permits. In other words, the amount of flow in the river can be lower and pumping will still be allowed. The review also found an inconsistency between permits on where the bypass flow is to be measured. The intent of this effort is to standardize and make consistent the bypass requirements for the remaining water users.

In order to amend a qualification on an existing water right permit, formal action by the SD Water Management Board is needed. In the next few weeks, you can expect to receive official notification scheduling a hearing before the Water Management Board. Again this does not pertain to amending your permit, only junior permit holders. If you wish to participate in the hearing you can, but you are not required to. There will be instructions with the notification on how to participate in the hearing if you so choose.

If you have any questions concerning this effort, please contact Mark Rath of the Water Rights Program at phone (605) 773-3352.

Sincerely,

for Jeanne Goodman, Chief Engineer
Water Rights Program



**DEPARTMENT of ENVIRONMENT
and NATURAL RESOURCES**

JOE FOSS BUILDING
523 EAST CAPITOL
PIERRE, SOUTH DAKOTA 57501-3182

denr.sd.gov

February 12, 2018

43-3, 519-3, 519-3A

Northern States Power Company
414 Nicollet Mall
Minneapolis MN 55401

Subject: Bypass Qualifications on Big Sioux River Water Rights

Dear Sir:

The SD Department of Environment and Natural Resources (DENR) has evaluated the minimum flow bypass condition on water rights that pump from the Big Sioux River. The Chief Engineer is recommending amending the bypass qualification on 21 Big Sioux River water rights. The intent is to bring uniformity to the bypass requirement placed on water rights in past years. The water right you own or operate does not have a bypass qualification and will not be amended by the proposed action.

Even though your water right is not being amended, DENR wants to keep you informed of the proposed action since it involves the Big Sioux River. Enclosed is a notice to be published on or before February 15, 2018, in one newspaper from each of the seven counties that the Big Sioux River flows through. Also, the staff report including affected water rights and the proposed changes is available at <http://denr.sd.gov/public> or contact Mark Rath at 605 773-3352 or mark.rath@state.sd.us.

A hearing before the Water Management Board is scheduled at 11:30 AM, Wednesday, March 7, 2018 in the Matthew Training Center, Joe Foss Building, 523 E Capitol Avenue, Pierre SD. The enclosed notice provides information on the hearing and ability to file a petition in either support or opposition to the proposed amendment. The filing of a petition must be postmarked by February 26, 2018. Additionally filing a petition affords you the opportunity to participate in the March 7th hearing before the Board if you choose.

Again, this does not pertain to amending your water right, only junior water right holders with existing bypass requirements. If you have questions, please call. DENR is willing to work to answer any questions.

Sincerely,

Eric Gronlund
Water Rights Program, DENR
605 773-3352
Eric.gronlund@state.sd.us

enclosure