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TEST PROGRAM ELEMENT II BLANKET AND SHIELD THERMAL-HYDRAULIC AND THERMOMECHANICAL TESTING, EXPERIMENTAL FACILITY SURVEY

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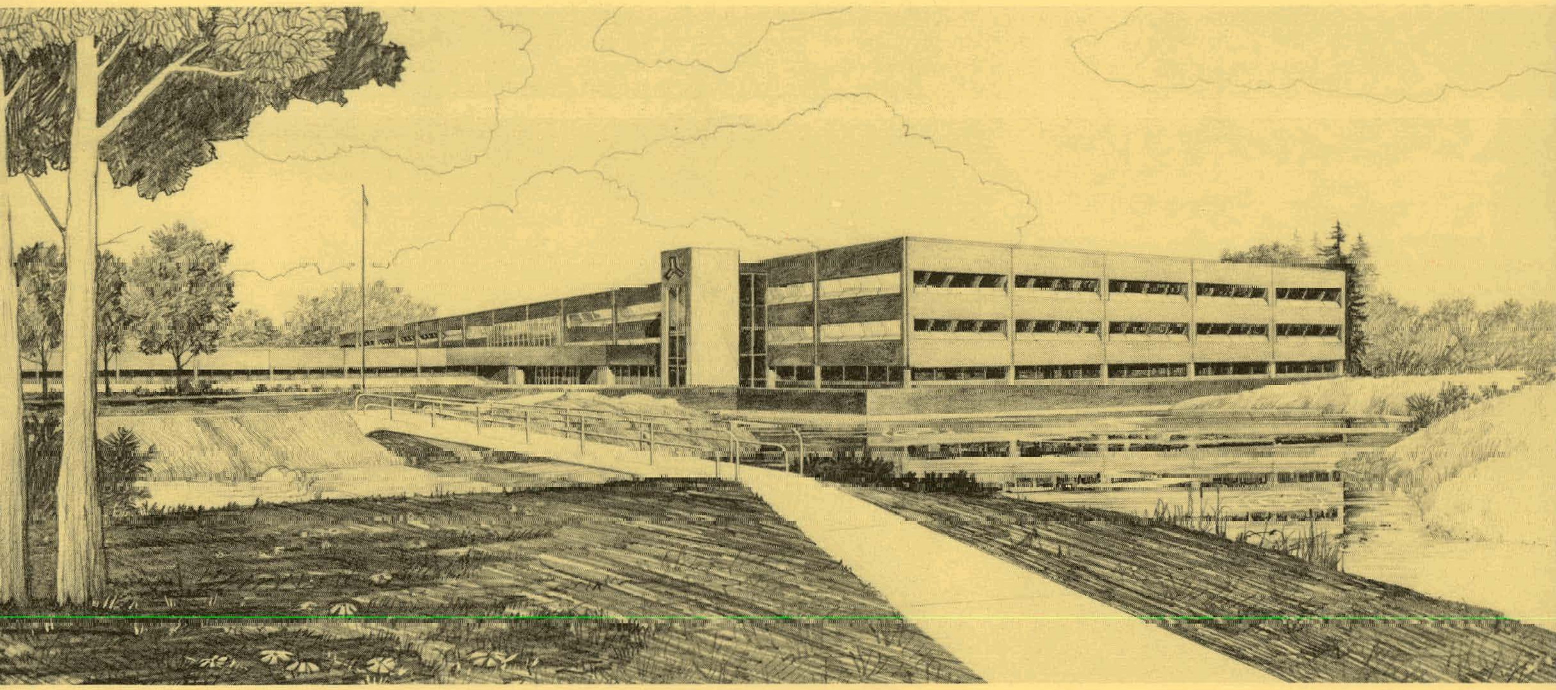

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ABSTRACT

This report presents results of a survey conducted by EG&G Idaho to determine facilities available to conduct thermal-hydraulic and thermomechanical testing for the Department of Energy Office of Fusion Energy First Wall/Blanket/Shield Engineering Test Program. In response to EG&G queries, twelve organizations (in addition to EG&G and General Atomic) expressed interest in providing experimental facilities. A variety of methods of supplying heat is available.

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TPE-II Facility Survey

1. Introduction

The Department of Energy (DOE) Office of Fusion Energy First Wall/Blanket/Shield Engineering Test Program under the direction of Argonne National Laboratory (ANL) includes Test Program Element-II (TPE-II) blanket/shield thermal-hydraulic and thermomechanical testing. Preliminary concepts for the program were submitted to ANL jointly by EG&G Idaho, Inc., and General Atomic Company (GA)[1,2,3] in competition with other corporations. Subsequently, EG&G and GA were contracted to conduct a planning study for the program.

The main objective of TPE-II is a thermal-hydraulic and thermomechanical test program which will be used to establish a data base for the design of fusion reactor blanket/shield assemblies and design verification tests of the resulting blanket/shield design. Parts of the planning stage for these tests include recommended concepts for testing and facilities available for conduct of that testing. This report presents a survey conducted by EG&G of available facilities. This survey will be used in coordination with the other test planning activities to decide where the testing should be undertaken.

2. FACILITIES SURVEYED

In order to determine the availability of testing facilities, major universities, corporations, and other institutions throughout the United States were contacted. A form letter soliciting information on candidate facilities for the test program was prepared. Enclosures to the letter outlined the test objectives and posed specific questions as to the capabilities of the testing facilities. A copy of the survey letter and questionnaire are included as Appendix A to this report.

Twenty-three prospective test organizations (other than EG&G/GA) were sent questionnaires. Of these, twelve expressed interest in providing facilities for at least a part of the TPE-II testing. A list of those contacted and their positive or negative interest is included as Table 1.

TABLE 1. RESPONSE TO TPE-II FACILITIES SURVEY

	Test Facilities Contacted	Response
1.	Hanford Engineering Development Laboratory Richland, WA	Yes
2.	TRW Defense & Space Systems Group Redondo Beach, CA	No
3.	Battelle Pacific Northwest Laboratories Richland, WA	Yes
4.	Boeing Engineering & Construction Company Seattle, WA	Yes
5.	Westinghouse Electric Corporation Pittsburgh, PA	Yes
6.	Combustion Engineering Inc. Windsor, CT	Yes
7.	Babcock & Wilcox Lynchburg, VA	Yes
8.	Columbia University New York, NY	Yes
9.	McDonnell Douglas Astronautics Company St. Louis, MO	Yes
10.	Lawrence Livermore National Laboratory Livermore, CA	Yes
11.	WYLE Laboratories Norco, CA	No
12.	University of Colorado Boulder, CO	No
13.	UCLA Los Angeles, CA	No
14.	Massachusets Institute of Technology Cambridge, MA	No

15.	NASA Lewis Research Center Cleveland, OH	No[a]
16.	Hughes Research Laboratories Malibu, CA	No
17.	Sandia Laboratories Albuquerque, NM	No
18.	Oak Ridge National Laboratories Oak Ridge, TN	Yes
19.	Argonne National Laboratory Argonne, IL	Yes
20.	Los Alamos National Laboratory Los Alamos, NM	Yes
21.	University of Missouri Columbia, MO	No
22.	Battelle Columbus Laboratories Columbus OH	No
23.	General Electric Company Schenectady, NY	No

a. Replied that no facilities were available.

3. SUMMARY OF EXPRESSIONS OF INTEREST

Responses from interested facilities were prepared in varying formats. Some answered the questions in the EG&G survey directly, some provided excerpts from their response to the original TPE proposal, while others enclosed company brochures as part of the response. Two approaches were selected to present this material. In this section an overview of the types of facilities is presented based on some of the more important testing capabilities listed in the EG&G questionnaire (in Appendix A). These cannot of course represent the full information contained in the responses, so the major portions of each response are presented in Appendices B through N of this report. Facility descriptions from the responses were included, but detailed drawings and photographs, of which many were received, were omitted.

Section 3.1 presents a general overview of the responding organizations' capabilities. Table 2 is a list of abbreviations of organizations used in this report. In many cases the responding organization would be able to provide a variety of test facilities, so an attempt was made to summarize the range of capabilities.

The capabilities can be divided into three general categories, based on the method of supplying heat. The first is by a fluid medium such as water, sodium, lithium, or helium. The second and third methods are by nonnuclear energy beams and nuclear radiation, respectively. Section 3.2 lists each fluid medium, organizations that can provide a facility using that medium, and a brief statement of the capabilities of each. Section 3.3 lists the types of energy beam sources, organizations that have such facilities available, and a brief statement of the capability of each. Section 3.4 describes potential nuclear test reactor heating sources.

3.1 General Overview

<u>HEDL</u>

There are five test loops available in which heated fluid flows through a piping system. Two loops are water and one each is sodium,

TABLE 2. ABBREVIATIONS FOR COMPANIES IN THIS REPORT

HEDL	Hanford Engineering Development Laboratory
PNL	Battelle Pacific Northwest Laboratories
BEC	Boeing Engineering and Construction Company
WEC	Westinghouse Electric Corporation
CE	Combustion Engineering
B&W	Babcock and Wilcox
CU	Columbia University
MDAC	McDonnell Douglas Astronautics Company
LLNL	Lawrence Livermore National Laboratory
ORNL	Oak Ridge National Laboratories
ANL	Argonne National Laboratory
LANL	Los Alamos National Laboratory
EG&G	EG&G Idaho
GA	General Atomics

lithium and helium. The power sources use electrical resistances which heat the water or liquid metal.

PNL

Four test loops are available in which heated fluid is passed through a piping system. Three loops use water as the heating medium, while the fourth uses helium. The power supplies are electrical resistance.

BEC

Thirteen vacuum/inert gas chambers are available in which energy can be beamed at the target source. Radiation sources are Xenon arc lamps.

WEC

Westinghouse has a large number of water filled flow loops available. Two liquid lithium loops are available as well as liquid sodium loops. High surface heat flux test facilities using plasma arc, laser, and electron beam heating also have potential applications.

CE

Three water loops that operate at PWR reactor conditions are available. Heating is accomplished by electric resistance methods.

B&W

One test loop is available in which heated water flows through a piping system into a large bay area. The heating method is electrical resistance.

CU

Two test loops in which heated water flows through a piping system are available. Heating by the electrical resistance method is used.

MDAC

Three vacuum/inert gas chambers are available in which energy is beamed at a target. Heating sources are quartz/tungsten lamps, graphite heaters, and a 15 kw CO_2 cw laser.

LLNL

One test chamber is available in which energy is beamed at the target. The heat source is a beam containing neutrals and ions of deutrium. Cooling is supplied by low pressure water.

ORNL

Available facilities include a large helium circulating facility, a large vacuum chamber, and water circulating loops. Heat sources are neutral beam injectors, small plasma torches, direct electrical, and radiant. ORNL also has several fission test reactors.

ANL

Argonne has several water filled loops available as well as one lithium loop and eight sodium loops. Several test reactors are also available.

LANL

A variety of thermal-hydraulic facilities is available. In addition, a small pool test reactor and a beam dump for beam heating devices are available.

EG&G/GA

EG&G has available eight loops with water as the medium, one loop with air/water, two loops with water/steam, two loops with helium, and one sodium and one lithium loop. Fission test reactors are capable of supplying peak fast neutron fluxes of up to 5×10^{15} cm^{-2} sec^{-1}.

GA facilities include a test facility to circulate hot helium through blanket modules. Also available are chambers in which specimens can be bombarded with ion beams, and can be subjected to high magnetic fields.

3.2 Fluid Heating Capabilities

This section summarizes the capabilities of organizations which can provide facilities for circulating heated fluids through piping systems. Categories for helium, lithium, sodium, and water are included. The general method of supplying energy is through electrical resistance heating; however, EG&G, ORNL, LANL, and others can provide test reactors in which nuclear heating takes place.

3.2.1 Helium

HEDL - 1.8 kg/s flow rate
3.2 MPa pressure
349°C[a] temperature
0.5 MW H/EX capacity

PNL - 182 kg/hr flow rate
2.17 MPa pressure
1150°C temperature
120 kW ac power supply

ORNL - 4 MW Power
10.5 MPa (1550 psi) pressure
600°C temperature
3.2 kg/s flow rate

---

[a] Could be increased by adding a heat exchanger.

LANL - 25 MPa pressure
one inch line

EG&G - 34.6 MPa pressure
1649°C temperature
Small flow rate

GA - two flow loops
625°C temperature
305 m/s flow rate

3.2.2 Lithium

HEDL - 37.8 l/s flow rate
0.34 MPa pressure
427°C[a] temperature
Power supply electrical heaters

WEC (LLP) - 45 liters lithium
260°C temperature
Rotating disk simulates high velocities
recirculation loop

WEC (LIFE) - 550°C temperature
7.6 l/s flow rate

ANL (LPIL) - 227 liter capacity
360 to 500°C temperature

EG&G (ARA-2) - 749°C temperature
0.0001 MPa pressure
Pool Boiling Reflux

---

[a] Can be upgraded to 540°C or more

3.2.3 Sodium

HEDL - 37.8 l/s flow rate
2 MPa pressure
650°C temperature
1.4 MW/18 kA
5 MW H/EX capacity

WEC (GPL-1) - 1.0 MW power
12.6 l/s flow rate
650°C temperature
2.37 MPa pressure

WEC (GPL-2) - 1.26 MW power
126 l/s flow rate
650°C temperature
1.96 MPa pressure

WEC (TSTF) - 650°C temperature
12.6 l/s flow rate
2.16 MPa pressure
1 MW power

ANL (SGTF) - 500 kw heater capacity
4.2 l/s flow rate
0.69 MPa design pressure
538°C design temperature

ANL (CCTL) - 60.6 l/s flow rate
4182 l capacity
649°C design temperature
0.45 MPa pressure

ANL (HTSL) - 3.8 l/s flow rate
200 kw power[a]
130 kw heat exchange at 649°C inlet
0.7 MPa pressure
538°C to 649°C temperature

ANL (HT-LMMHD) - 38 l/s flow rate
- 0.69 MPa pressure
- 538°C temperature

ANL (AT-LMMHD) - 25 l/s flow rate
- 2.0 MPa pressure
- 53°C temperature

ANL (CAMEL-II) - Trace heating
538°C temperature
2.07 MPa pressure
270 l capacity
16.7 l/s flow rate

ANL (OPERA) - Trace heating
649°C temperature
1.38 MPa pressure
295 l capacity
19 l/s flow rate

ANL (SSL) - Trace heating
375 kw heat exchanger
649°C temperature
3.24 MPa pressure
9.1 l/s flow rate
225 l capacity

---

[a] Upgrading to 1 MW

EG&G (TRA) 1.1 MPa pressure
704°C temperature
0.7 l/s flow rate

3.2.4 Water

HEDL (Hydromechanical) - 946 l/s flow rate
1.72 MPa pressure
120°C temperature
Variable DC power supply

HEDL (High pressure) - 15.8 l/s flow rate
1.72 MPa pressure
315°C temperature
4 MW/32 kA power supply

PNL (High pressure heat transfer) - 15.8 l/s flow rate
15.27 MPa pressure
324°C temperature
4 MW and 1.25 MW dc power supplies

PNL (High pressure low heat transfer) - 6.3 l/s flow rate
13.9 MPa pressure
315°C temperature
35 KW ac* power supply

PNL (Hydraulic) - 44 l/s flow rate
3.5 MPa pressure
121°C temperature
75 kW ac[a] power supply

WEC (TMHL) - 9.55 l/s flow rate
1.38 MPa pressure
82°C temperature
200 kW power supply

---

[a] Up to 4MW by rerouting power supply

WEC (MPHL) - 378 l/s flow rate
1.38 MPa pressure
82°C temperature
200 kW power supply

WEC (Boiling HT) - 2.5 kg/s flow rate
0.88 MPa pressure
121°C temperature

WEC (A&B Loops) - 9.5 l/s flow rate
16.5 MPa pressure
343°C temperature

WEC (D Loop) - 278 l/s flow rate
16.5 MPa pressure
343°C temperature

WEC (E Loop) - 126 l/s (at 0.45 MPa) flow rate
62 l/s (at 0.78 MPa) flow rate

WEC (PWRSD) - 0.6 l/s flow rate
343°C temperature
15.5 MPa pressure
200 kW power supply

WEC (J Loop) - 28.4 l/s flow rate
343°C temperature
17.2 MPa pressure
3.5 MW power supply

WEC (H Loop) - 883 l/s flow rate
93°C temperature

CE (TF-1) - 25.2 l/s flow rate
15.8 MPa pressure
343°C temperature
300 kW dc or ac programmable[a] power supply

CE (TF-2) - 94.6 l/s flow rate
17.2 MPa pressure
343°C temperature

CE (TF-16) - 9.46 l/s flow rate
17.2 MPa pressure
343°C temperature
250 kW* power supply

B&W - 41 l/s flow rate
21 MPa (3000 psi) pressure
370°C (700°F) temperature
10 MW/50 kA power supply

CU - 53.6 l/s flow rate
343°C temperature
11.5 MW steady state power
50 Ka power supply
16.5 MPa pressure

ANL (SGTF) - 12.5 l/s flow rate
16.55 MPa pressure
482°C temperature

ANL (FIVTF) - 60.5 l/s total flow rate
1.03 MPa pressure
50°C temperature

---

[a] 2.17 MW steady state power

ANL (500-gpm loop) - 37.9 l/s flow rate
0.28 MPa pressure

ANL (MCTF) - 151.5 l/s flow rate
37.8°C temperature

ANL (OPMTF) - 30.3 l/s flow rate
0.21 MPa pressure
93°C temperature

LANL - Four test cells
40 ft tall high bay
2.5 MW power supply
500 KW dc power supply

EG&G (Ambient flow calibration) - 2.1 MPa pressure
52°C temperature
0.017 m/s flow velocity

EG&G (MTR/ETR test loop) - 1.8 MPa pressure
65°C temperature
0.038 m/s flow velocity

EG&G (Ballistic flow calibration) - 0.8 MPa pressure
65°C temperature
0.076 m/s flow velocity

EG&G (Shim rod loop) - 3.2 MPa pressure
115°C temperature
0.006 m/s flow velocity

EG&G (ATR Safety Rod Drive) - 3.2 MPa pressure
115°C temperature
0.05 m/s flow velocity

EG&G (HT Hydraulic test loop) - 7.7 MPa pressure
260°C temperature
0.076 m/s flow velocity

EG&G (Air/Water) - 2.2 MPa pressure
93°C temperature
1.41/0.05 m/s flow velocity

EG&G (ARA High temperature) - 15.6 MPa pressure
343°C temperature
0.019 m/s flow velocity

EG&G (Fast Loop) - 7.8 MPa pressure
279°C temperature
0.303 m/s flow velocity

EG&G (Blowdown facility) - 15.6 MPa pressure
288°C temperature
0.32 m^3 vol blowdown

EG&G (Two-phase flow loop) - 7 MPa pressure
284°C temperature
0.568 m/s flow velocity

3.3 Beam Heating Capabilities

In contrast to the facilities which supply heat via a flowing fluid, several of the responding organizations can supply heat energy by beaming the energy at the target.

3.3.1 Lasers

WEC - CO_2 Laser
25 kW
flux at 640 MW/m^2 over 1 mm dia
38 MW/m^2 over 1 cm dia
continuous or pulsed

MDAC - CO_2 Laser
up to 1 x 1 meter
12.5 kW max output power
10 kW/cm^2 max flux

3.3.2 Neutral Beam

LLNL - Neutrals and ions of deutrium
Power of 4.6 MW
Max flux of 9 kW/cm^2
80 keV

ORNL - 60 keV
0.9 MW x 2 injectors

3.3.3 Lamps

BEC - Xenon arc
35 kW power

MDAC - Quartz/tungsten
to 3000°C
Max power 50 W/cm^2

3.3.4 Other

WEC - SF electron beam
- plasma arc
- ion beam

MDAC - Graphite heaters

ORNL - Small plasma torches

ANL - Electrically heated tungsten filament
300 kW
3410°C
Argon gas environment

GA - Superconducting high field ion beam

EG&G - Electrically heated tungsten mesh furnace
530 kW
3000°C (vacuum), 2700°C (Helium)
Four separately heated and cooled segments

LANL - Vacuum chambers (which can be used with radiant heater panels or other energy sources)

3.4 Nuclear Heating Capabilities

As a separate activity from the questionnaire in response to which the preceding data were furnished, a survey was conducted of nuclear test reactors which may be considered for use in TPE-II. Table 3 summarizes the findings of that survey.

Several existing test reactors have one or more in-core tubes 15 cm or greater in diameter which would be suitable for testing of module subassemblies or small diameter cylindrical modules such as the ORNL/Westinghouse concept. Reactors of this type which, in addition, have suitable flux levels are listed in Table 4.

TABLE 3. REACTOR TEST VOLUME SIZES

REACTOR	POWER[1] (MW)	EXP. NEUTRON FLUX[1] ($n/cm^2/sec$)	EXP. SIZE[1] (cm)	TEST POSITION EXP. VOL. (P)	NOTES
EBR-II (Idaho)	62.5	2×10^{15}F	7.4 Dia x 36	1.5	NA cooling-in core dia.
HFBR (New York)	40	5.5×10^{14}Th 2.0×10^{14}F	2.4 Dia x 7.6	0.03	Central, peripheral and reflector locations
ATR (Idaho)	250	5.3×10^{14}F	1.5 Dia x 122	0.2	12 A-Holes
		1.9×10^{14}F	2.2 Dia x 122	0.5	8 B-Holes
		3.8×10^{14}F	1.6 Dia x 122	0.2	16 H-Holes
		3×10^{12}F	12.5 Dia x 122	14.9	4 I-Holes
		3×10^{12}F	8.1 Dia x 122	6.2	16 I-Holes
		2.8×10^{11}F	45.3 x 56.6 x 122	312.8	All O-Holes together
		1.5×10^{15}F	7.6 Dia x 122 (9)*	5.6	Flux traps * Thru hole capability
ETR (Idaho)	175	4×10^{14}F	7.6 Sq x 91+	5.3	+ thru hole capability
		4×10^{14}F	15.4 Sq x 90+	21.7	* Modification required for large blanket
		4×10^{14}F	15.4 x 22.9 x 91+	32.2	
		4×10^{14}F	22.4 Sq x 90+	47.9	
		3.4×10^{14}F	65 x 76 x 91*	451.3	
LOFT (Idaho)	50	N/A	No test positions	--	
PBF (Idaho)	28	2×10^{13}F	15.5 Dia x 91	17.2	At 28 MW/Max power 270GW Integrated power 1350MW-sec
OMEGA-W (New Mexico)	8	5×10^{13}Th	5.1 Dia x 671	1.2	MTR type core
MITR (Mass)	4.9	3×10^{13}Th	4.5 Dia x 61 (2)	0.9	Vertical thimbles
		1×10^{14}F	2.9 Dia x 61 (2)	0.3	
			6.9 x 5.1 x 61	2.1	

TABLE 3. (continued)

REACTOR	POWER[1] (MW)	EXP. NEUTRON FLUX[1] ($n/cm^2/sec$)	EXP. SIZE[1] (cm)	TEST POSITION EXP. VOL. (P)	NOTES
BSR (Tenn.)	2	5.5×10^{12}Th	7.6 x 7.67 x 61	3.5	Also cryogenic facility
BFIR (Tenn.)	100	1.3×10^{15}F	13 Dia x 51*	6.8	* Maximum dia. of flux trap
		9×10^{14}Th	1.3 Dia x 51 (8)	0.1	
		2×10^{14}Th	3.8 Dia x 5 (11)	0.6	
		1.5×10^{14}Th	7 Dia x 51 (2)	2.0	
ORR (Tenn.)	30	4.5×10^{14}F	7.8 Sq x 38.4	2.3	Any core position
		4×10^{13}F	71 x 76 x 63	339.9	Pool side irradiation
MURR (Missouri)	10	4.6×10^{14}Th 3.5×10^{13}F	3.8 Dia x 75 (3)	0.9	Flux trap positions
FFTF (Washington)	400	4.5×10^{15}F	7 Dia x 91 (2)	3.5	Closed loops
			11 Dia x 91 (8)	8.6	Open loops, IN core dia. Any fuel or reflector Position
GETR (California)	50	3.3×10^{14}Th	7.4 Dia x 91 (1)*	3.9	In standby condition
		2.8×10^{14}Th	7.4 Dia x 91 (2)*	3.9	* thru loop capability
		2.7×10^{14}Th	3.8 Dia x 91 (8)	1.0	
BR2 (Belgium)	50	6×10^{14}Th 2.4×10^{15}F	20.3 Dia x 91	29.5	Center hole
		2.8×10^{14}Th 9.4×10^{14}F	20.3 Dia x 91 (4)	29.5	Thru holes
		2.8×10^{14}Th 9.4×10^{14}F	5 Dia x 91 (10)	1.8	Several other irr. positions Fluxes given are Max. Values

TABLE 3. (continued)

REACTOR	POWER[1] (MW)	EXP. NEUTRON FLUX[1] ($n/cm^2/sec$)	EXP. SIZE[1] (cm)	TEST POSITION EXP. VOL. (P)	NOTES
DR-3 (Denmark)	10	1×10^{14} Th	17.8 Dia x 61	15.2	
HFR (Netherlands)	20	2×10^{14} Th 5×10^{14} F	14.5 Dia x 60	9.9	Thru hole
		1.5×10^{14} Th 4×10^{14} F	6 Dia x 60	1.7	In-Core U-Tube
SAFARI-1 Thru loop capability	20	4×10^{14} Max Th		15 x 15 x 61	13.7

Sources of Information:

1. "Research, Training, Test, and Production Reactor Directory", American Nuclear Society, First Edition, (1979).

2. "Directory of Nuclear Reactors", Vol. 1-4 AIEA (1959).

The PBF Reactor has been included in this list because of its unique power pulse capability. During a disruption, 200 to 400 Joules/cm^2 of charged particle energy is dumped to the first wall surface in an estimated 20 msec. The PBF can be pulsed in this time scale to 270 GW with an integrated power of 1350 MW-sec. By using a ^{3}He surface heating converter concept, greater than 200 Joules/cm^2 of charged particle energy could be dumped to the module surface.

TABLE 4. HIGH FLUX REACTORS CAPABLE OF CONDUCTING 15 cm DIAMETER IPT LOOP EXPERIMENTS

Reactor	Location	Number of Test Holes (>15 cm dia)	Fast Flux in Test Volume (n/cm^2/sec)
ETR	INEL	5	4×10^{14}
PBF	INEL	1	2×10^{13}
BR-2	Belgium	5	2.4 to 9.4×10^{14}
DR-3	Denmark	1	1×10^{14}
SAFARI-1	South Africa	1	4×10^{14}

In-core test facilities, as described above, are not sufficiently large to permit testing of larger modules which basically present large plane surfaces to the plasma source. To accomplish this, it is necessary to position the module adjacent to one of the faces of the core. Two existing test reactors listed in Table 5 have this physical capability. Possibly others could be modified by the removal of their thermal columns.

Fusion oriented nuclear test facilities have not been considered here. Such facilities as the Rotating Target Neutron Source (RTNS-II) and the Fusion Materials Irradiation Test (FMIT) Facility have intense sources of 14 MeV neutrons but their test volumes are not sufficiently large to allow meaningful thermal-hydraulic and thermomechanical testing. Near-term fusion reactors such as the Tokamak Fusion Test Reactor (TFTR) may be

TABLE 5. HIGH FLUX REACTORS WITH PLANE SOURCE CAPABILITY

Reactor	Test Vol. (L)[a]	Fast Neut. Flux $n/cm^2/sec$	Fission Reactor Equiv. Neutron Wall Loading for Bulk Heating (MW/m^2)	Time Required for Life-time Fluence (yr)
ETR	450	3.4×10^{14}	1.8	1.2
ORR	340	4.0×10^{13}	0.2	10.5

a. Test volume calculated for a 63 cm thick blanket.

considered, but the neutron wall loading ($0.1\ MW/m^2$) is so much lower than the 1-4 MW/m^2 required for realistic heating studies, and the burn time (~ 1 s) is so short, that the usefulness of such devices for this kind of testing appears to be lacking. If an INTOR class machine is built, it may provide a means of performing thermal-hydraulic or thermomechanical testing on fusion blanket modules, but special heating augmentation techniques such as those described in Reference 4 would be required.

4. CONCLUSIONS

Twelve organizations, other than EG&G and GA, have expressed interest in providing facilities for TPE-II testing. Of these, some such as B&W and CU can provide a single specialized capability, while others such as WEC and EG&G offer a wide range of facilities available.

Two primary heating methods have been addressed--fluid heating and energy beam heating. Each of these groups has been divided into subgroups. In addition, potential sources of nuclear heating were discussed.

The recommended method for selecting facilities is to first decide on the type of test and the method of supplying heat. Then Section 3 of this report can be used to quickly determine which facilities can provide such heating or loading. A brief summary of the capabilities of each is presented. Finally, with the list so narrowed, the more comprehensive submittals in Appendices B through N can be used to study the facilities in more detail. This task would no doubt need to be followed up by more contacts with the responding organizations to fill in the details imposed by the specific test requirements. At this point in the overall plan, a decision might be made as to the most appropriate test facility for the particular experiment.

References

1. L. F. Burdge (EG&G) to R. E. Tiller (DOE) Letter LFB-40-79, "Information for the First Wall, Blanket and Shield Verification Testing Program", July 13, 1979.

2. P. Y. S. Hsu et al., "Fusion Technology Development--First Wall/Blanket System and Component testing in Existing Facilities", EGG-FT-5281, December 1980.

3. Technical Proposal GACP 11-118, "Test Program Element II, Blanket and Shield Thermal-Hydraulic and Thermomechanical Testing of the Office of Fusion Energy First Wall/Blanket/Shield Engineering Test Program", May 11, 1981.

4. G. A. Deis et al., "Evaluation of Alternative Methods of Simulating Asymmetric Bulk Heating in Fusion Reactor Blanket/Shield Components," EGG-FT-6503, October 1981, EG&G Idaho, Inc., Idaho Falls, ID.

APPENDIX A

EG&G QUESTIONNAIRE
FOR EXPRESSION OF INTEREST
IN TPE-II TESTING

EG&G Idaho, Inc.
P.O. BOX 1625, IDAHO FALLS, IDAHO 83415

September 11, 1981

THERMAL-HYDRAULIC THERMOMECHANICAL TESTING FACILITIES

Dear

EG&G Idaho is conducting a survey of test facilities suitable for thermal-hydraulic and/or thermomechanical testing of fusion blanket/shield components. This survey is being conducted as part of Test Program Element-II (TPE-II) of the DOE Office of Fusion Energy First Wall/Blanket/Shield Engineering Test Program. This letter is a request for information concerning test facilities under your control or to which you may have access which may be suitable for such testing.

A testing program is envisioned under TPE-II which will address test objects, test conditions and performance parameters such as those listed in Enclosure 1. The purpose of the test program is to contribute to a thermal-hydraulic and thermo-mechanical data base for the design of fusion reactor blanket/shield assemblies by conducting experiments on actual or simulated blanket/shield components. These experiments will pinpoint specific needs for further research and provide data on concepts for blanket/shield designs. The information collected as part of this survey will be published in a report to Argonne National Laboratory. They are directing the test program. That program is expected to begin in approximately six months and to run for two years with possible extensions.

Would you please respond to the questions of Enclosure 2. We would welcome specification sheets, layout drawings, flowsheets, etc. Additionally, we would be pleased to discuss your response by telephone.

We request that you respond by September 25, 1981. If you have questions, please call Dr. Glen Longhurst at (208) 526-9950 or FTS 583-9950.

Very truly yours,

P. Y. Hsu
TPE-II Program Manager
Fusion Technology Program

TEST OBJECTS AND CONDITIONS TO BE ADDRESSED IN TPE-II

The testing being planned for TPE-II will begin with fairly small-scale, separate-effects type tests and progress to large scale integrated system tests. Typical objects to be tested include:

(a) Monolithic stainless steel blocks with integral cooling channels.

(b) Stainless steel/boron carbide composite blocks with integral cooling channels.

(c) Facsimiles of modular solid breeder canisters with integral cooling channels.

(d) Facsimilies of modular liquid metal cooled canisters with integral liquid metal flow paths.

(e) Advanced versions of objects (a) through (d) above outfitted with manifolds, connectors, and support pieces.

(f) Versions of objects (a) through (e) above connected in series or in parallel with one or more additional objects of the same type.

Testing conditions sought will be representative of anticipated blanket/shield environments for near term fusion machines. These conditions will vary depending on the blanket/shield design being tested but may include:

(1) Simulated assymmetric bulk (nuclear) heating up to 40 w/cc (peak heating rate) for normal condition testing and up to 100 w/cc for transient condition testing. [For objects (a) through (f) above.]

(2) Pressurized water coolant operating at coolant exit temperatures up to 300 C and coolant pressures up to 2000 psi. [For objects (a), (b), (c), and, where appropriate, (e), and (f) above.]

(3) Pressurized helium coolant operating at coolant exit temperatures up to 600 C and coolant pressures up to 1000 psi. [For objects (c) and, where appropriate, (e) and (f) above.]

(4) Liquid metal coolant operating at coolant exit temperatures up to 450 C and pressures up to 500 psi. [For objects (d) and, where appropriate, (e) and (f) above.]

(5) Peak structural temperatures that are consistent with recommended upper limit values based on existing materials performance data.

Response of test pieces to both normal and off-normal or transient conditions will be investigated. The relationships between the parameters and relationships shown in Table 1 will be sought under each condition as shown.

Table 1

DESCRIPTION OF FW/B/S PERFORMANCE PARAMETERS AND RELATIONSHIPS TO BE ADDRESSED AS PART OF TPE-II

Performance Feature	Operating Condition: Normal	Operating Condition: Transient
Performance Parameters		
- Coolant Temperature Profile	X	X
- Test Object Temperature Profile	X	X
- Coolant Pressure Drop	X	X
- Coolant Velocity Profile	X	X
- Operational Heat Transfer Coefficient	X	X
- Local Strains and Stresses	X	X
- Deformations/Expansions	X	X
- Vibration Characteristics	X	X
- Flow Redistribution		X
- Temperature Redistribution		X
Performance Relationships		
- Steady-State and Cyclic Operation	X	X
- Failure Mode	X	X
- Sensitivity to Geometry	X	X
- Power Transients		X
- Partial Flow Blockage		X
- Loss of Coolant Flow/Pressure		X
- Adjacent Module Failure		X

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- Local Strains and Stresses	X	X
- Deformations/Expansions	X	X
- Vibration Characteristics	X	X
- Flow Redistribution		X
- Temperature Redistribution		X
Performance Relationships		
- Steady-State and Cyclic Operation	X	X
- Failure Mode	X	X
- Sensitivity to Geometry	X	X
- Power Transients		X
- Partial Flow Blockage		X
- Loss of Coolant Flow/Pressure		X
- Adjacent Module Failure		X

QUESTIONNAIRE FOR TPE-II FACILITY SURVEY

1. What facilities do you have for conducting steady or cyclic thermo-mechanical heating experiments?

   a. What is the heat source?

   b. What is the test environment (vacuum, inert gas, etc.)?

   c. What size and configuration of test pieces will they accommodate?

   d. What materials can be accommodated in the test space?

   e. What range of temperatures and/or power can be achieved?

   f. What are the rise and decay times of temperature/power in these facilities?

   g. What capability exists for active cooling of a test piece in these facilities?

   h. How are these facilities instrumented?

   i. What is the availability and approximate operating cost of these facilities?

   j. What supporting facilities are available, e.g., machine shops, analytical laboratories, etc.?

   k. What are the numbers and qualifications of personnel available to support tests in these facilities?

   l. What other aspects of these facilities would be pertinent to tests of this type?

2. With respect to thermal-hydraulic test facilities, please answer (a) through (l) above and the following additional questions.

   a. What is the configuration of the facility (circulation loop, blowdown, two-phase, etc.)?

   b. What fluids can be used? (H_2O, gas, liquid metal, etc.)

   c. What heat sources and sinks are available?

   d. What flow rates, pressures, pressure drops, velocities, etc., can be attained?

3. What capability do you have for providing nuclear heating?

   a. What neutron flux is attainable?

   b. What is the size and configuration of the test volume?

   c. How uniform is the neutron flux within that volume?

   d. What are the rate and the distribution of gamma heating within the test volume?

   e. What is the availability and cost of using these facilities?

   f. What is the environment (medium, temperature, pressure, etc.) of the test space?

   g. What supporting facilities are available?

   h. What supporting personnel are available?

2. With respect to thermal-hydraulic test facilities, please answer (a) through (l) above and the following additional questions.

   a. What is the configuration of the facility (circulation loop, blowdown, two-phase, etc.)?

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   b. What is the size and configuration of the test volume?

   c. How uniform is the neutron flux within that volume?

   d. What are the rate and the distribution of gamma heating within the test volume?

   e. What is the availability and cost of using these facilities?

   f. What is the environment (medium, temperature, pressure, etc.) of the test space?

   g. What supporting facilities are available?

   h. What supporting personnel are available?

APPENDIX B

APPENDIX B

HANFORD ENGINEERING DEVELOPMENT LABORATORY RESPONSE

Hanford Engineering Development Laboratory

P.O. BOX 1970 RICHLAND, WA 99352

September 18, 1981

Dr. P. Y. Hsu
TPE-II Program Manager
Fusion Technology Program
EG&G Idaho, Inc.
P. O. Box 1625
Idaho Falls, ID 83415

Dear Dr. Hsu:

As you may know, HEDL responded to the call for expression of interest (EOI) for TPE-II. Much of what you requested in your September 11 letter is contained in the EOI, which I understand is not available to you through Argonne. We are forwarding appropriate sections of the EOI for use in your survey.

The material provided is in skeleton form. Should you require additional information, please feel free to call. One area not covered sufficiently in our TPE-II proposal is the experiment support facilities at HEDL. These facilities are used for design, fabrication and testing of a wide variety of nuclear tests. Most of the efforts involve in-reactor instrumented test vehicles. Because of this, HEDL has developed excellent facilities for fabricating and working with physical measurement apparatus including thermocouples, heat pipes, transducers, accelerometers, flowmeters, heaters, strain measuring devices, and pressure measuring devices to name a few.

In addition to in-reactor tests, the experimental support laboratories provide technical backup to FMIT and RTNS-II. We believe that our test support facilities are unequaled elsewhere in the nuclear community.

J. J. Holmes, Manager
Components Technology

sgs

Enclosure

APPENDIX A

HYDRO-MECHANICAL TEST LOOPS AND POWER SUPPLIES

FOR TPE-II

HYDRO-MECHANICAL FACILITIES TO BE USED IN TPE-II TESTING

Facility	Coolant	Flow	Pressure	Temperature	Purity Control	Power Supply	Hex Capacity	Owner	Custodian
Hydromechanical (1)	Water	0-15000 gpm	0-250	70-250°F	PH	-	Feed & Bleed 300 gpm	DOE	HEDL
High Pressure	Water	0-250 gpm	0-2500 psig	70-600°F	PH, O_2	4 MW 32K amp	4 MW	DOE	PNL
Transient Test Loop	Sodium	0-600 gpm	0-300 psig	300-1200°F	Cold Trap <1 ppm O_2	1.4 MW 18K amp	5 MW	DOE	HEDL
Experimental Lithium System (2)	Lithium	0-600 gpm	0-50 psig	400-800°F (4)	Cold Trap Hot Trap <100 ppm N_2	-	(3)	DOE	HEDL
He Gas Loop	Helium	0-4 lb/sec	0-465 psia	420-660°F (5)	None	-	0.5 MW	DOE	HEDL

GENERAL NOTES: 1. Under power supply heading, these are controllable DC supplies. These supplies are capable of transient power cycling.

2. Future expected use of these facilities is low enough that no scheduling conflicts are expected which would prevent running substantial FW/B/S tests in the time period proposed.

(1) Addition of a shell and tube hex at low cost would extend heat dissipation to any desired level.

(2) Addition of a heat exchanger is required to match heat input.

(3) Existing heat removal may have to be increased.

(4) Can easily upgrade to at least 1000°F.

(5) Limit applies to He compressor. Higher test section outlet temperatures can be obtained by adding a heat exchanger downstream of test section.

A P P E N D I X B

PERSONNEL FOR TPE-II ACTIVITIES

PHASE I WORK

WAYNE L. THORNE
MANAGER, HYDRAULICS & MECHANICS
BSME OREGON STATE UNIVERSITY 1957

Mr. Thorne worked in fluid mechanics and heat transfer applied development in support of Hanford Production Reactors, the Plutonium Recycle Test Reactor, Canadian CANDU Reactors, and Hanford's N Dual-Purpose Reactor for ten years with the General Electric Company. Full scale and modeled electrically heated simulated fuel columns were used in pressurized water cooling experiments to determine maximum allowable operating conditions. Heat fluxes to 10 MW/M^2 coolant temperatures to 635°F and pressures to 2500 psig were used. During his five years with Battelle's Pacific Northwest Laboratory, heat transfer and fluid mechanics experiments in the above areas continued plus similar experiments, under contract, for Westinghouse PWR's. Mr. Thorne was manager of Experimental Thermal Hydraulics for Battelle. During Mr. Thorne's ten years with the Hanford Engineering Development Laboratory, work has included design, construction, and operation of a 600 gpm Liquid Sodium System with capability for using electrically simulated fuel elements for steady state and transient heat transfer studies; design, construction, and operation of a 600 gpm Liquid Lithium System in support of Fusion Materials Irradiation Testing; and management of Hydraulics and Mechanics (stress, vibration, shock, and seismic loadings) in support of LMFBR, Fusion and Base Technology programs.

DEAN R. DICKINSON, PRINCIPAL ENGINEER

PhD, Chemical Engineering, University of Wisconsin, 1958

Dr. Dickinson has 23 years of experience in research, development, and testing in the areas of corrosion, reactor coolant systems, heat transfer, and fluid mechanics. He has been responsible for planning test programs in these areas, performing the tests, and interpreting the results. Heat transfer experience has included extensive work with high-pressure water and tests with electrical resistance heating producing surface heat fluxes up to 5 MW/m^2. Other experience potentially applicable to FW/B/S problems include spray cooling of hot metal surfaces, scale deposition to water under high heat flux and radiation, gas cooling of simulated irradiated nuclear fuel assemblies, and in-reactor coolant and corrosion testing.

J. A. RYAN
FELLOW ENGINEER

Mr. Ryan's present assignments are primarily in the areas of applied and experimental mechanics. He is responsible for the application of computer-aided-design techniques to in-house experimental high temperature test loops. Additionally, the analytical areas have included elastic and inelastic analysis of components subjected to thermal shock. Mr. Ryan also directs the activities of the vibration test and analysis group. The testing includes modal testing as well as flow-induced vibration testing, data acquisition and analysis.

FREDERICK R. FISHER, ADVANCED ENGINEER
MSEE University of Michigan (1967)

Mr. Fisher worked on design and specification of nuclear data acquisition equipment and data analysis, including computer interfacing and application, during three years with Battelle's Pacific Northwest Laboratory. During the succeeding nine years at Westinghouse Hanford, he developed thermal, hydraulic, and mechanical data acquisition systems for reactor models and prepared acceptance test procedures and design modifications for the High Temperature Sodium Facility process control and monitoring instrumentation. Current assignments include software development for thermal-hydraulic testing and hardware and software development for advanced ultrasonic inspection techniques.

JAMES L. STRINGER, SENIOR ENGINEER

BSEE, Oklahoma State University (1956)
MSEE, University of Washington (1966)

Mr. Stringer worked on the development of nuclear radiation detection process and dosimetry instrumentation systems for general plant use and specific facilities during his nine years with General Electric Company. During his five years with Battelle's Pacific Northwest Laboratory in the Applied Physics Group, he participated in the development of a liquid surface ultrasonic holography system, as well as high energy pulsed laser systems. During this time he led the development of an ultra-low noise charge sensitive amplifier for use with high resolution gamma energy detectors. During the past 10 years he has been with the Hanford Engineering Development Laboratory where significant contributions have been made in both process and special instrumentation systems used in liquid sodium systems. In the past three years he has participated in the successful development of four sensors which have been installed in the central LOFT reactor fuel bundle. These sensors which include measurements of temperatures to 4000°F (2200°C), pressures to 2500 psig and displacements to 1-inch were required to meet accuracies of 2% to 3% of reading while being subjected to environmental transients which exceeded 600°F per minute and pressures to 2500 psig. The successful completion of this program required the development of special materials and processes to meet the size and performance requirements. Mr. Stringer holds 4 patents and has authored numerous publications.

APPENDIX-C

APPENDIX C

BATTELLE PACIFIC NORTHWEST LABORATORIES RESPONSE

Battelle
Pacific Northwest Laboratories
P.O. Box 999
Richland, Washington U.S.A. 99352
Telephone (509) 375-2873
Telex 15-2874

September 18, 1981

P. Y. Hsu
TPE-II Program Manager
Fusion Technology Program
EG&G
P. O. Box 1625
Idaho Falls, Idaho 83415

SUBJECT: THERMAL-HYDRAULIC THERMOMECHANICAL TESTING FACILITIES

Dear Mr. Hsu:

As per your request in a September 11, 1981 letter to D. E. Olesen, we have attached a listing of facilities operated by Battelle Pacific Northwest Laboratories that could be used to perform separate effects test on fusion reactor first wall/blanket/shield components. This listing was previously prepared as part of PNL's response to TPE-I and should answer most of your questions. If you have questions concerning any of these facilities, please call Jim Creer (509) 375-2664 or myself (509) 375-2873.

Sincerely,

M A McKinnon

M. A. McKinnon
Senior Research Engineer
Energy Systems Engineering Section

MAM:fo

Attachment

8.0 EXPERIMENTAL AND OTHER FACILITIES (TOPIC H)

PNL and BEC have selected thermal-hydraulic and thermal-mechanical facilities that could be used to perform selected FW/B/S tests. The PNL facilities are owned by DOE and there are minimal facility use charges to DOE-sponsored projects. The major cost associated with the experimental facilities is the charge for power consumption.

The use of these existing DOE test facilities would permit separate effects testing (e.g., hydraulic characterization, critical heat flux, etc.) to be initiated during the second year of the project. Data obtained from these early experiments would start the formation of a data base for follow-on testing and for analytical model development. Adequate analytical models or design analysis capabilities are necessary for planning and performing larger combined effects and integrated effects tests.

If PNL and BEC were selected to perform TPE-I, fusion work requiring use of the PNL facilities identified in the following sections would receive first priority. Other projects requiring use of these facilities and BEC facilities would be coordinated with the schedule of the FW/B/S project so that TPE-I milestones will be met. Past experience and the projected future use of these facilities indicate that the progress of the FW/B/S testing project will not be affected by other projects.

8.1 PNL FACILITIES

Four major thermal-hydraulic experimental facilities that can be used to perform FW/B/S testing are briefly described below. Discussions of other equipment and facilities such as data acquisition systems, computer facilities, and instrumentation calibration laboratories are also presented.

8.1.1 High Pressure Heat Transfer Facility

The high pressure heat transfer facility is a stainless steel recirculating loop capable of supplying high pressure single phase water or two-phase steam-water to test pieces under investigation. The following loop operating conditions can be obtained:

- 250 gpm @1450 ft head
- 2200 psig
- 615^{o}F

- 4 MW and 1.25 MW dc power supplies

Both horizontal and vertical test pieces can be accommodated in the two test section areas.

A 1.25 MW (50 v; 22,500 amps) dc power supply (two motor generators) is normally used to supply power to a preheater to obtain desired inlet test piece coolant temperatures. Automotically controlled power ramps can be obtained, and ramps of 0-90% over 1-2 seconds are possible. The 4 MW (125 v; 32,000 amps) silicon rectifier power supply is usually used to provide power to a test piece. Automatically-controlled power ramps over 1-100 seconds can be produced. Steep power ramps of 0-90% in 1-2 sec can be provided. The electrical buss system is designed to permit either power supply to be used for either test piece or preheater power.

The facility has been used extensively for performance of thermal-hydraulics experiments related to nuclear technology. The following list identifies typical studies performed in the facility in the past:

- Critical Heat Flux
- Single and Two-Phase Pressure Loss
- Thermal Mixing
- Flow Instability
- Transient Heat Transfer
- Spray Cooling
- Fretting-Corrosion

The past studies on critical heat flux, flow instability, and fretting-corrosion are similar to those that could be performed as part of the FW/B/S test project.

8.1.2 High Pressure Low Flow Heat Transfer Facility

A second high pressure stainless steel facility will also be available to the FW/B/S test project. This loop is smaller and does not have the flow capacity of the facility presented in the previous section. Typical operating conditions are:

- 100 gpm @180 ft heat
- 2000 psig
- 600^{o}F
- 35 kW ac power supply

The buss system of the 1.25 and 4 MW power supplies could be rerouted to provide more power to the high pressure low flow facility.

This facility has been used to perform such studies as nuclear fuel pin rupture characteristics, film boiling heat transfer, and electrically simulated FFTF fuel pin instrumentation integrity tests. The loop control console is presently being upgraded to permit unattended operation.

8.1.3 Hydraulic Facility

The hydraulic facility is constructed of carbon steel and can be operated as a once-through or recirculating system. The loop is equipped with four pumps that can be combined to provide the following operating conditions:

- 700 gpm @ 1150 ft head (1800 gpm @210 ft head)
- 500 psig
- $250^{\circ}F$
- 75 kW ac power supply

The buss system of the 1.25 and 4 MW power supplies could be rerouted to provided more power to this facility.

The loop is equipped with a laser Doppler anemometer (LDA) to permit local measurements of velocity and turbulence intensity. The LDA system and other temperature, pressure, and flow instrumentation have resulted in the following types of studies being conducted in this facility:

- Single phase pressure loss
- Hydraulic characterization (velocity, turbulence intensity) of rod bundles
- Hydraulic characterization of rod bundle spacers
- Flow distributions near blocked bundles
- Natural, forced, and combined convection flow in rod bundles

Flow characterization of FW/B/S test pieces could be investigated using the hydraulic facility if this degree of detail becomes necessary.

The hydraulic facility is also equipped with two NBS-traceable flow meter calibration loops. The low flow calibration loop can be used to calibrate flowmeters in the 0.1 gph-100 gpm range. The high flow calibration loop can be used up to 1100 gpm.

8.1.4 High Temperature Helium Loop

A high temperature helium loop is available for investigating gas coolant heat transfer. The following operating conditions can be obtained with the loop:

- 400 lb/hr
- 300 psig
- 2100°F
- 120 kW ac power supply

Two test section zones are provided for test piece evaluation. One test section is oriented horizontally, and the other vertically. The two test sections can be used separately or simultaneously.

The loop is equipped with a gas purification/monitoring system that will permit impurity control. The system will produce ultra-high purity helium with less than 1 ppm total impurities, including water.

8.1.5 Data Acquisition Instrumentation

Three types of data acquisition systems (DAS) can be used to support testing activities in the four previously discussed experimental facilities. They are:

- PDP-11/34 computer
- Monitor Lab 10091 DAS with magnetic tape
- Fluke 2240 DAS with magnetic tape

These systems supplement standard loop instrumentation and chart recorders found in each heat transfer facility.

8.1.6 Instrumentation Calibration Lab

PNL does not operate an instrumentation calibration laboratory at Hanford. However, the Hanford Engineering Development Laboratory (HEDL), operated for DOE by Westinghouse, does have an NBS-traceable laboratory. Essentially, all PNL instrumentation calibrations are performed in the HEDL standards lab. This eliminates the costly, time-consuming need to go offsite for calibration services. As mentioned in a previous section, one of the PNL facilities is equipped with NBS-traceable flowmeter calibration loops.

8.7.1 Large Computer Facilities

PNL has access to an onsite Univac 1100 operated by Boeing Computer Services for DOE. PNL can use the Brookhaven CDC 7600, the Los Alamos CDC 7600, and the University Computer Center (Dallas, Texas) Univac 1108 and CDC 6600 on DOE-sponsored projects, with special permission from DOE. Battelle-Northwest owns a VAX computer, which can also be used with permission from DOE.

APPENDIX D

APPENDIX D

BOEING ENGINEERING AND CONSTRUCTION COMPANY RESPONSE

BOEING ENGINEERING AND CONSTRUCTION COMPANY
P.O. Box 3707
Seattle, Washington 98124

A Division of The Boeing Company

October 2, 1981

Dr. P. Y. Hsu
TPE-II Program Manager
Fusion Technology Program
EG&G Idaho Inc.
P.O. Box 1625
Idaho Falls, Idaho 83415

Dear Dr. Hsu,

I apologize for the lateness of this response but it is my understanding that you are still in a position to consider capabilities for TPE-II tests.

The Boeing Company is equipped to support areas 1, and 2 of your questionnaire; namely, thermo mechanical heating and thermal-hydraulic testing. We are not, however, presently equipped to perform in the area of nuclear heating.

I am enclosing a summary of our relevant capabilities which were submitted as part of an earlier response to TPE-II. You will note that the radiation sources are Xenon arc lamps. The test environments we can provide are versatile including vacuum, and inert gas. Our test chambers will accommodate very large test pieces. We have also small chambers which can be more economically employed for smaller items. We have a comprehensive range of machine shops, analytical laboratory support, and personnel skills. We are prepared to provide approximately costs in response to a more detailed inquiry.

Although not specified in the enclosed information, we believe we have the necessary equipment to provide for the thermal hydraulic testing and would be pleased to address your specific needs.

If you have further questions, do not hesitate to call.

Yours truly,

Bryan Cairns

R. Bryan Cairns
Manager
Advanced Technologies

Enc

BOEING

BOEING ENGINEERING AND CONSTRUCTION COMPANY

CAPABILITIES

- High Flux Test Facilities
- Space Environment Simulation Laboratory

HIGH FLUX TEST FACILITIES:

DESCRIPTION AND APPLICATIONS

2-5615-8000-053

PREPARED BY THE SOLAR/THERMAL RADIATION LABORATORY

A. R. LUNDE
773-8516

STEVEN DURICK
773-8516

BOEING AEROSPACE COMPANY
Kent Space Center
P.O. Box 3999, M/S 86-01
Seattle, Washington 98124

GENERAL PURPOSE SODIUM LOOPS (GPL)

The General Purpose Loops No. 1 and No. 2 (GPL-1, GPL-2) were commissioned in 1967 and 1971, respectively. These high-temperature sodium test facilities were designed, as their names imply, as general loops for utilization in a wide range of concurrent test programs. This approach deviates from the "one facility - one program" concept, resulting in fuller utilization of both facilities and manpower on LMFBR Programs.

Both GPL's have employed the "parallel test section" approach over their operational life. Facility design is such that test articles can be installed, removed, or serviced in these parallel test section areas while the facility continues operation on other on-going programs. The result is full facility utilization with optimum economic benefits to the customer. Facility operating costs to be incurred by any one program can be minimized by (1) sharing operating costs with concurrent programs, or (2) operating with more than one test section.

In both GPL's, the basic facility has been expanded to include parallel sodium test circuits, resulting in increased versatility. The results have been to increase the versatility of the facility. The GPL-1 facility is now a two-loop system, and GPL-2 is three-loop. The respective facility expansions identified above provide facility versatility by (1) providing increased flow and power capability for future test programs, and (2) permitting parallel, concurrent operation of the various systems at different operating conditions. An additional capability that the multiple-loop feature brings to the GPL-1 and GPL-2 facilities is the ability to impose thermal transients and thermal shocks on test articles through the use of circulating sub-loops at different sodium temperatures.

Because of the inherent time-consuming efforts associated with facility start-up and shutdowns, both facilities are operated on a 24-hour day, continuous basis. Actual continuous operating periods for the two facilities have ranged from one week to six months.

Both facilities are currently in use on the following programs:

Facility	Program
GPL-1	Radial Blanket Heat Transfer Test Program
GPL-1	LMFBR Thermal Striping Evaluation Program
GPL-1	LMFBR Double-Wall Steam Generator Test Program
GPL-1	CRBRP Dynamic Friction (Seismic) Test Program
GPL-2	CRBRP Primary Shutdown Systems Program

In summary, ARD currently has two operational Sodium Test Facilities with capability for multiple, concurrent test program implementation. The upgrading and expansion over the years, including the recent addition of Distributed Data Acquisition Systems at both facilities, provide versatile "test beds" for future DOE programs. The application of the parallel test section concept permits maximum economic benefits to the customer; and the application of the "multiple-loop" concept provides facilities with full capabilities, including transient and thermal shock, for future DOE programmatic requirements.

OPERATING AND DESIGN PARAMETERS

	GPL-1 Facility		GPL-2 Facility		
	GPL-1	**GPL-1A**	**Loop #2**	**Loop #2A**	**Loop #2B**
Piping	2" Sch. 40 (304 SS)	2" Sch. 40 (304 SS)	6" Sch. 40 (304 SS)	3" Sch. 40 (304 SS)	3" Sch. 40 (304 SS)
Pump	EM Flat Linear Induction Pump	EM Flat Linear Induction Pump	EM Flat Linear Induction Pump	EM Annular Linear Induction Pump	EM Annular Linear Induction Pump
Heater	3.4×10^6 BTU/Hr Output	3.4×10^6 BTU/Hr Output	4.3×10^6 BTU/Hr Output	3.4×10^6 BTU/Hr Output	3.4×10^6 BTU/Hr Output
Flowrate	Up to 200 GPM	Up to 200 GPM	Up to 2000 GPM	Up to 300 GPM	Up to 300 GPM
Temperature	1200°F MAX	1200°F MAX	1200°F MAX	1100°F MAX	1100°F MAX
Pressure	330 psig MAX	330 psig MAX	270 psig MAX	270 psig MAX	270 psig MAX
Cold Trap Flow	2 GPM	2 GPM	5 GPM	5 GPM	5 GPM
Oxygen Purity	1.0 to 5.0 PPM	1.0 to 5.0 PPM	1.0 to 5.0 PPM	1.0 to 5.0 PPM	1.0 to 5.0 PPM
Carbon Purity	0.2 PPM MAX	0.2 PPM MAX	0.2 PPM MAX	0.2 PPM MAX	0.2 PPM MAX
Hydrogen Detection	ON-LINE HYDROGEN METER				
Sodium Chemistry	ON-LINE VANADIUM WIRE STATION				

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1.0 INTRODUCTION

1.1 General Information

The Solar/Thermal Radiation Laboratory has been active in developing state-of-the-art technologies for high flux testing since the 1960's. The High Flux Test Facility, shown in Figures 1 & 2, is a versatile Laboratory radiant heating system that closely duplicates the spectral content of the sun's rays. The facility is designed for solar power evaluation, thermal nuclear pulse testing, general environmental experiments and a variety of thermal balance studies. The facility has the capabilities of producing heat flux of 0.1 to 50.0 megawatts per square meter over a test article size of 10 to 4000 square centimeters. This facility is located at the Boeing Kent Space Center, Kent, Washington.

1.2 General Description

The High Flux Test Facility consists of a rotational containment vessel (±180°), support 'A Frame' structure and a mounting alignment structure for the XM-300 source modules. The XM-300 is a solar simulation instrument consisting of a water-cooled Xenon short-arc lamp, adjustable over a nominal input power plane of 7-35 kilowatts, a solid state D.C. power supply with suitable controls and monitors, and a collector system to transfer the energy from the source to the target plane. The operation of the XM-300 is entirely controlled from the control console, where the instrument may be turned "ON" and "OFF", the light source operating parameters may be monitored, and any malfunctions will be indicated. The XM-300 is internally air and water-cooled and has been designed to operate for extended continuous operation. The High Flux Test Facility utilizes four (4) XM-300 source modules.

1.3 Theory of Operation

1.3.1 Optical System

The XM-300 optical system collects and distributes the optical radiation from a water-cooled 20 kilowatt or 32 kilowatt Xenon Short-Arc lamp to illuminate a projected area. The optical system is a specially developed water-cooled source collecting mirror-mounted with its focus coincident with the arc of the lamp. The optical contour of the collector mirror surface is

1.3.1 Optical System (cont.)

an Aconic section designed to optimize the energy transfer from the arc of the lamp to the optical train of the system. The collected light is then directed toward the target plane.

1.3.2 Electrical System

The 20 kilowatt and 32 kilowatt lamps are powered by a LC-52 D.C., three phase, 100% duty cycle rectifier type D.C. power source. The D.C. power supply provides power to the starter, which yields a 60,000 volt pulse required to ignite the Xenon lamp, and directly to the lamp to substain the arc between the Anode and Cathode.

1.3.3 Containment Vessel

The containment vessel is an aluminum framed supporting structure (2.0m x 2.0m x 1.9m) which contains the test fixture supporting frame and the air-actuated beam douser paddle. This structure supports the four (4) XM-300 source modules and the adjustable test plane structure.

2.0 FACILITY APPLICATIONS

The High Flux Test Facility has a wide range of possible testing applications. The facility has the unique ability to either adjust the source modules alignment or adjust the test plane distance from the light source in order to obtain the desired flux levels and beam diameters. These adjustments allow the test parameters to vary greatly with respect to incident heat flux (0.1 to 50.0 megawatts/sq. meter) and to the test article's size (10 to 4000 sq. centimeters), (see Figure 3). Listed below are a few of the applications being tested or planned for testing with this facility.

2.1 Solar Central Receiver Cavity Insulation Evaluation

Several design concepts and types of insulation material were tested in the bench model solar receiver. The four generated light beams overlapped to produce a concentrated flux level between 550 and 600 kilowatts per square meter over a 30 centimeter diameter circle. The test specimen was fixed at 32° angle tilt above horizontal to simulate the gravitational effects within the actual bench model solar receiver. See Figure 4. The cavity set-up at

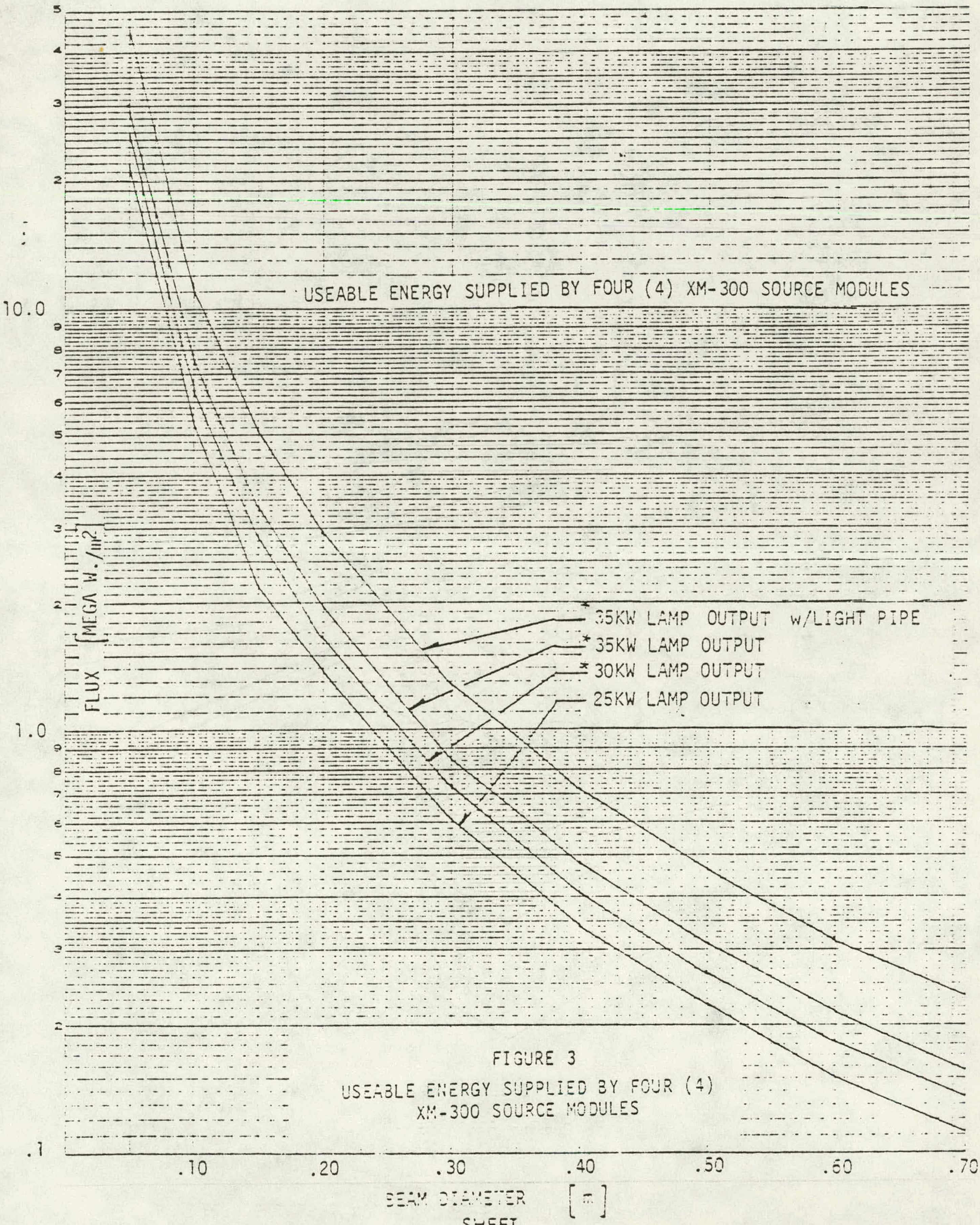


FIGURE 3

USEABLE ENERGY SUPPLIED BY FOUR (4) XM-300 SOURCE MODULES

* Based on future upgrading of system.

High Flux Solar Simulation Test Facility

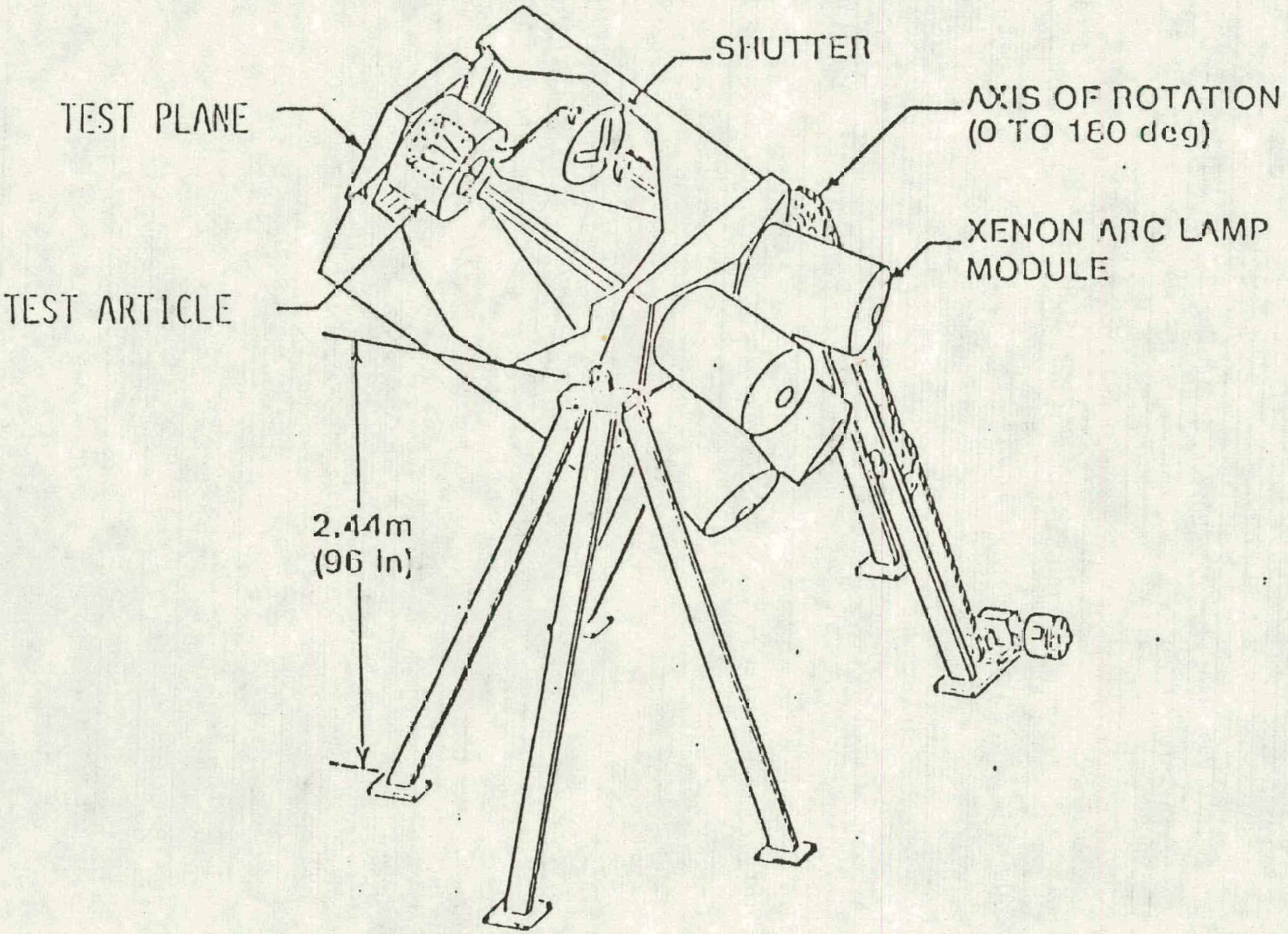


FIGURE 4
HIGH FLUX SOLAR SIMULATION TEST FACILITY

2-5615-800C

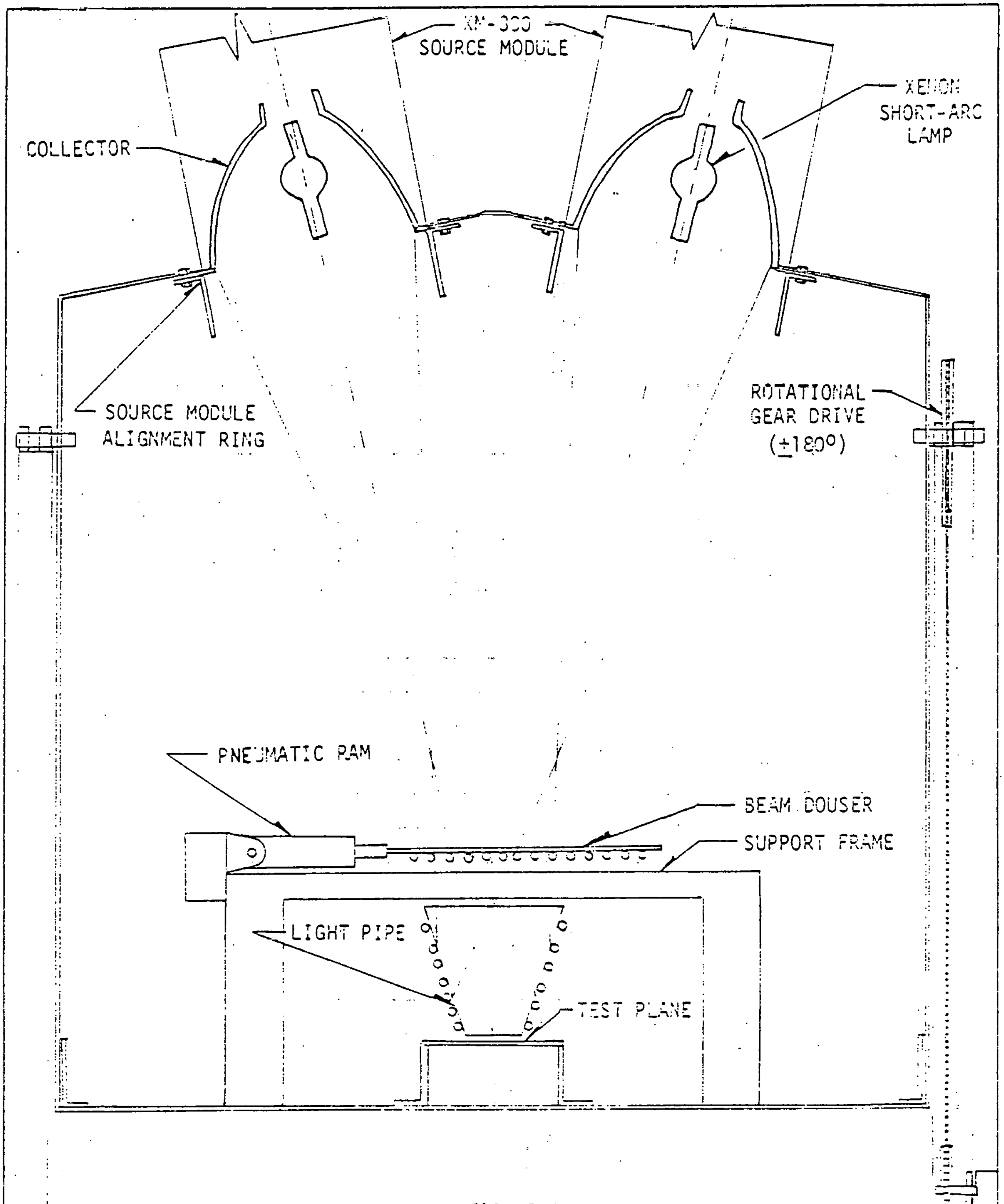


FIGURE 6

NUCLEAR THERMAL PULSE TESTING SCHEMATIC

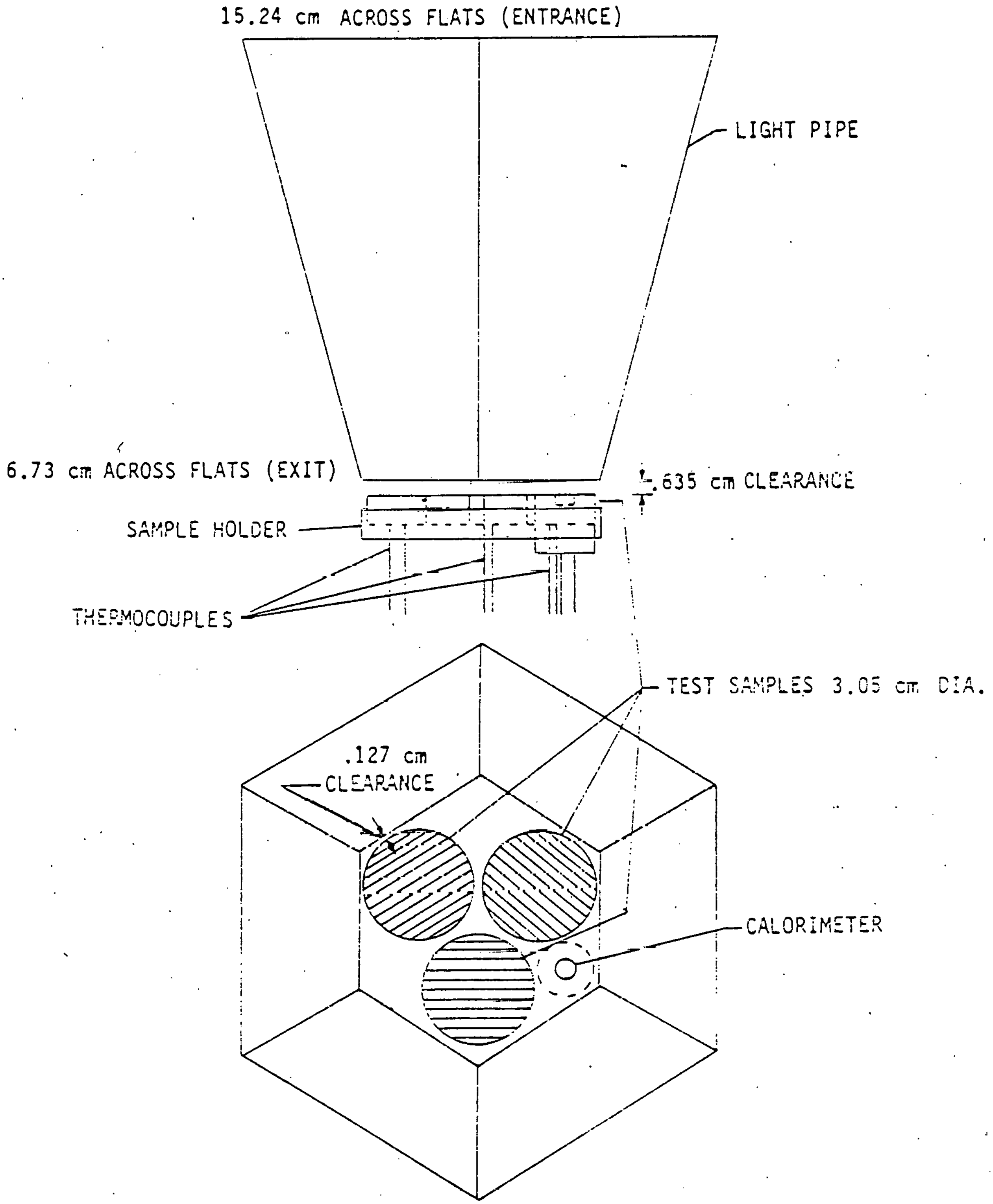


FIGURE 7
LIGHT PIPE SCHEMATIC

2.1 Solar Central Receiver Cavity Insulation Evaluation (cont.)

the test plane is shown in Figure 5. The maximum surface temperature of the sample before failure was 1760°C.

2.2 Nuclear Thermal Pulse Set-Up

2.2.1 General Description

The Nuclear Thermal Pulse Set-Up is used to simulate the thermal energy released by a nuclear detonation. A schematic of the test set-up is shown in Figure 6. The collector directs the energy from the lamp down through a light pipe (Figure 7) onto a test specimen. The purpose of the light pipe is to collect the energy and redistribute it uniformly over the target area. A special energy beam douser is mounted above the light pipe. This water-cooled douser with a special shaped aperture in the center is pneumatically controlled. The douser is initially positioned to block the energy beam. Then, on command, the douser passes the aperture to the beam centerline at a controlled rate. At this point a programmer takes over control of the power supplies and decreases the lamps power at a rate such that the best possible pulse width at half maximum irradiance and total incident energy required are achieved. The douser is then positioned to block the beam. See Figures 8 and 9 for photographs of test equipment and typical test set-up.

The uniformity of irradiance and peak irradiance are recorded using a X-Y plotter prior to the test. This is accomplished by scanning the energy beam with the douser open. The calorimeter is mounted on a scanner arm with a position-indicating potentiometer (Figure 10). The calorimeter and potentiometer are connected to the plotter, and the calorimeter is driven back and forth through the energy beam recording calorimeter output versus position. See Figure 11 for typical uniformity of irradiance curves.

The thermal shape is determined prior to test by recording the calorimeter output versus time at the beam center during the operation of the douser. (Figure 12). See Figures 13 and 14 for typical pulse shapes. Figures 11 through 14 were generated with only one XM-300 source module.

PERCENT OF MAX IRRADIANCE REQUIRED

100

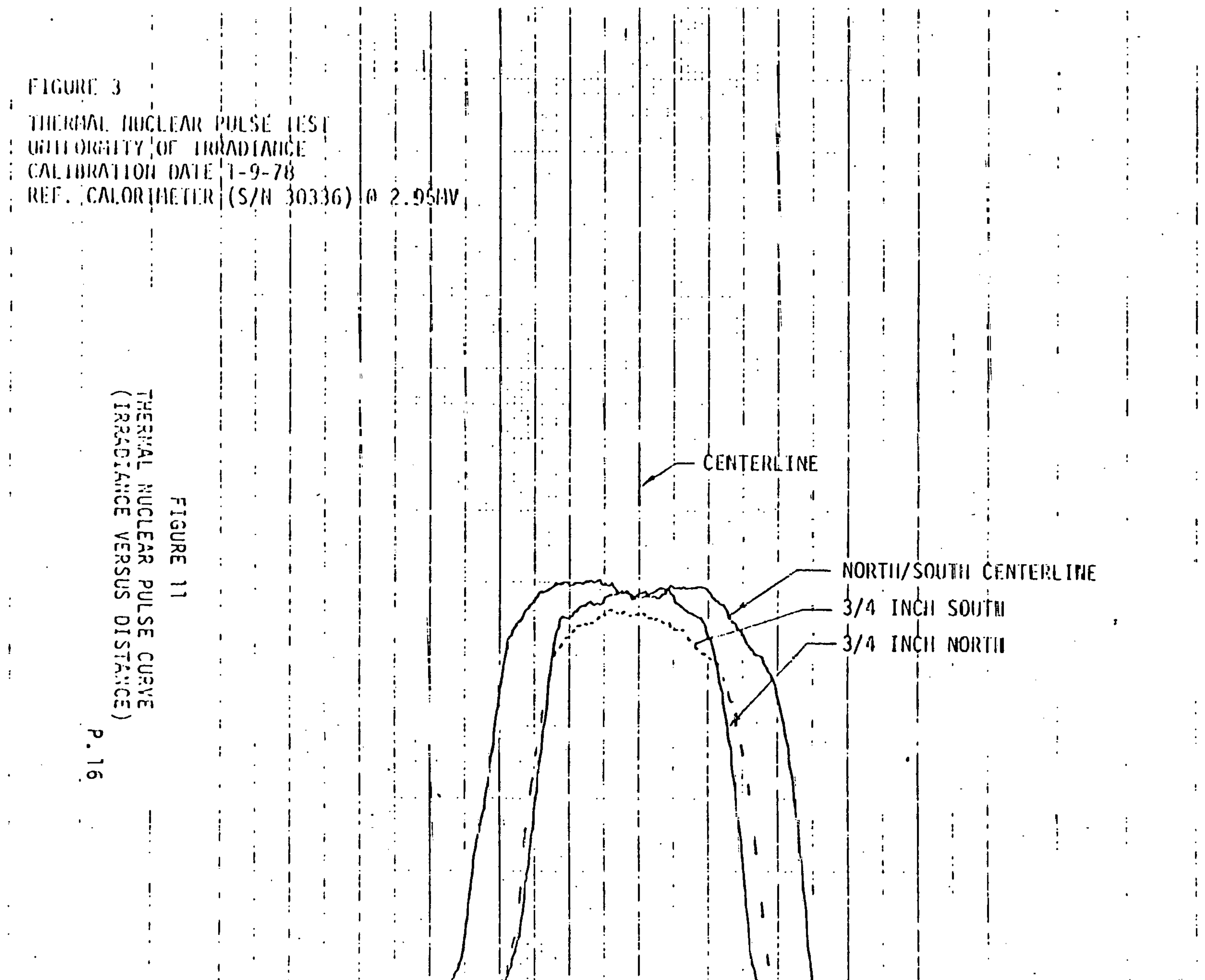
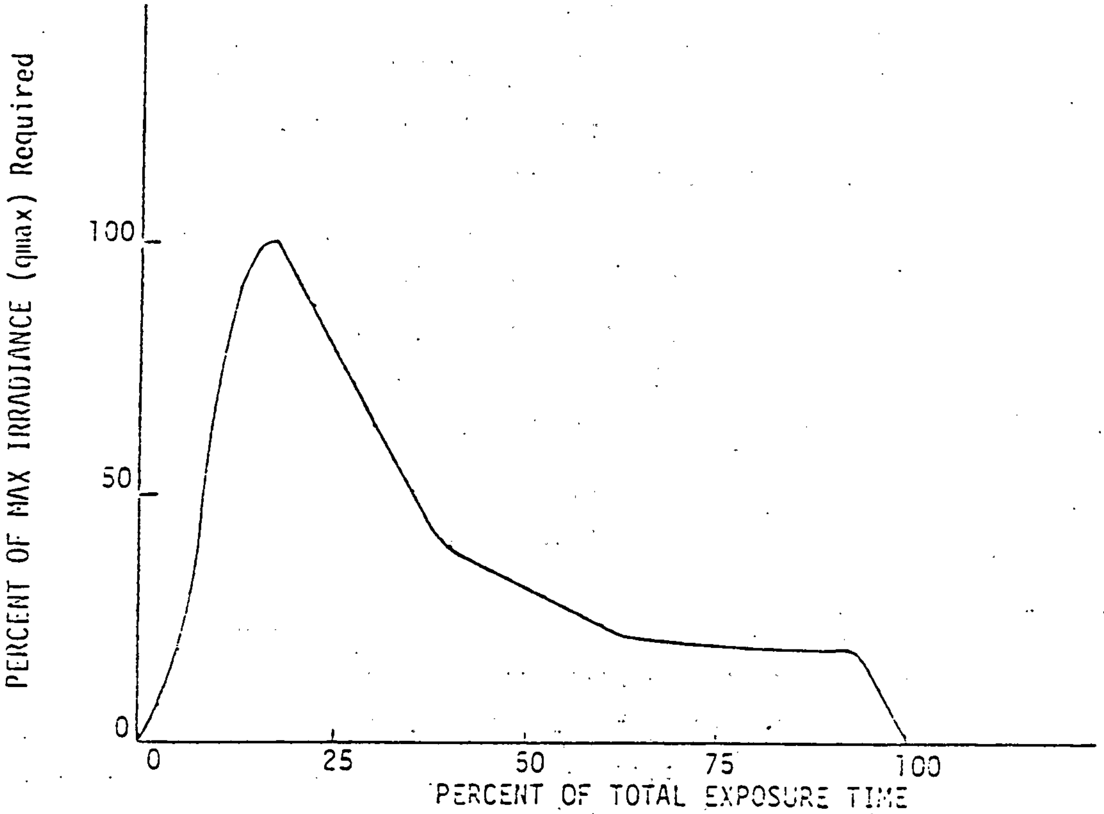


FIGURE 11
THERMAL NUCLEAR PULSE CURVE
(IRRADIANCE VERSUS DISTANCE)

THERMAL NUCLEAR PULSE
IRRADIANCE VS TIME

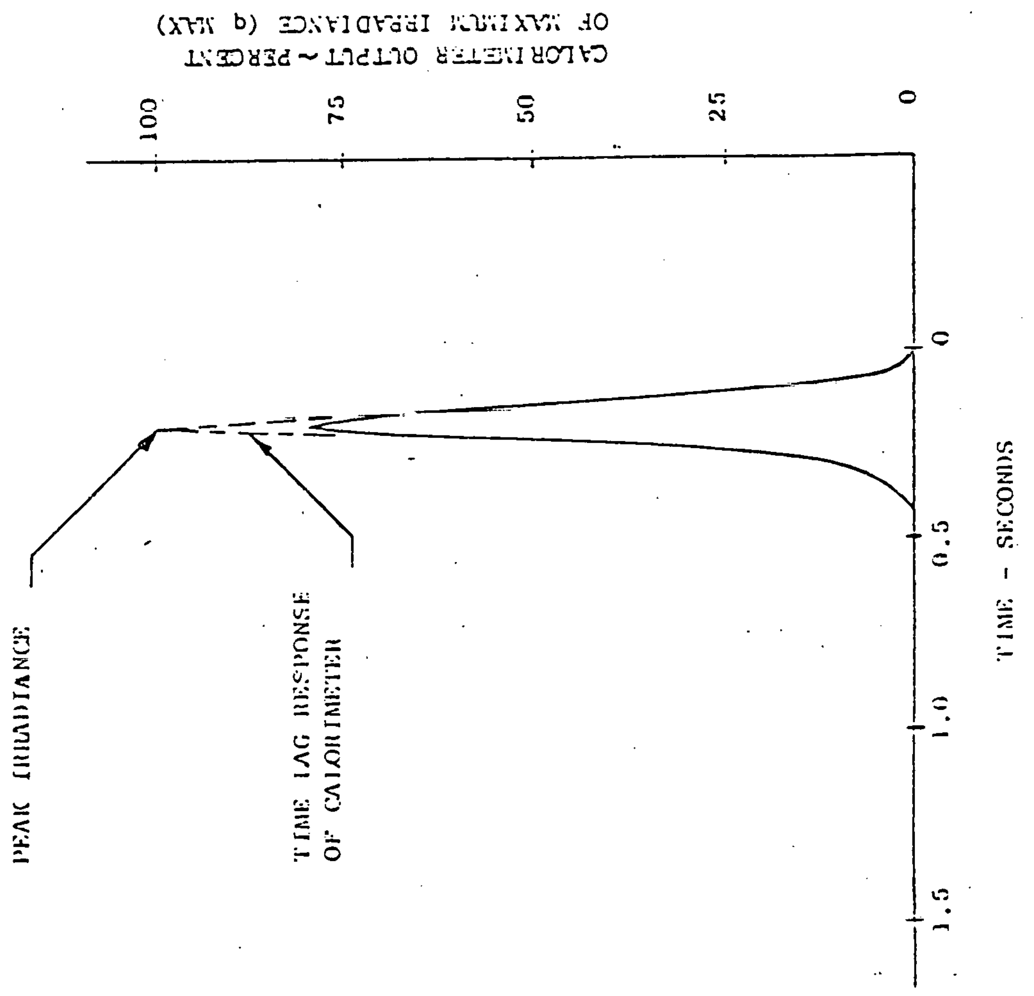


$(q_{MAX}) \times (\text{TOTAL EXPOSURE TIME}) = 108\ \text{Cal/cm}^2$

Q_{Total}/q_{max} = 9.11 sec. per pulse

Calibration Date: 1-13-78

FIGURE 12
THERMAL NUCLEAR PULSE CURVE
(IRRADIANCE VERSUS TIME)



THERMAL PULSE FACILITY
THERMAL PULSE SHAPE
TEST CONDITION - SET A
CALORIMETER:
HY-CAL MODEL C-1300A
S/N 30336
REF. STANDARD S/N 20045

DATE: 12-11-74
OPERATOR: A. R. LUNDE
PEAK IRRADIANCE @ 3.68 MV (REF)
Q TOTAL/q MAX = 0.104 SECONDS

FIGURE 13
TYPICAL THERMAL PULSE SHAPES

THERMAL PULSE FACILITY
THERMAL PULSE SHAPE
TEST CONDITION - SET B
CALORIMETER:
HY-CAL MODEL C-1300A
S/N 30336
REF. STANDARD S/N 20045

DATE: 12-9-74
OPERATOR: A. R. LUNDE
PEAK IRRADIANCE @ 0.835 MV
(REF) WITH NEUTRAL DENSITY FILTER
Q TOTAL/q MAX = 0.546 SECONDS

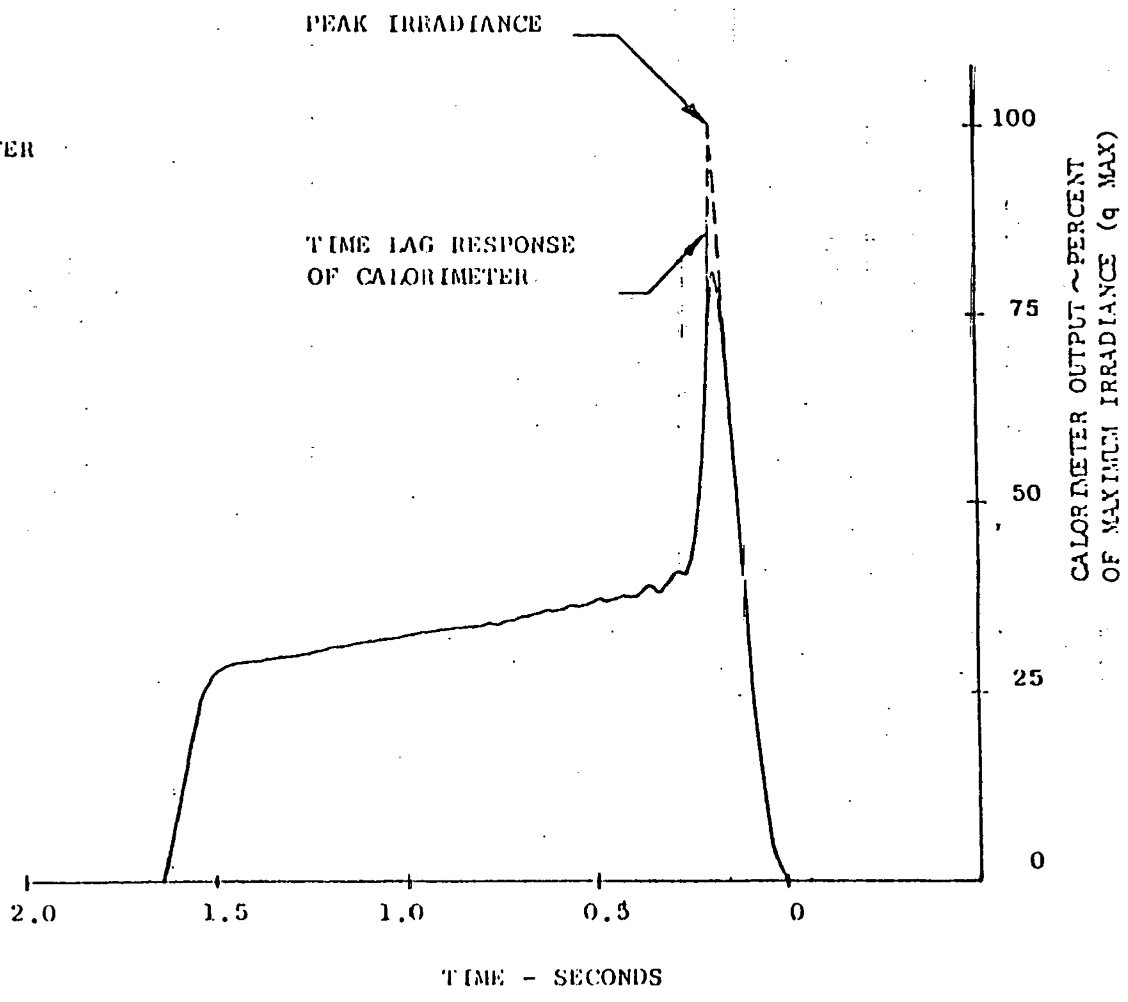


FIGURE 14
TYPICAL THERMAL PULSE SHAPES

2.2.2 Specifications

Irradiance: 0 to 3.4 x 10^6 W/M^2
(300 BTU/FT^2.Sec)
(80 cal/cm^2.Sec)

NOTE: 1) Capability of one XM-300 source module

2) Higher irradiance levels can be achieved with a new configuration light pipe with a decrease in beam size.

Uniformity of Irradiance: 15%

Beam Size: Hexagonal in shape -
6.73 cm (2.65 in.) across the flats

Spectral Energy Distribution:
.25 to .40 microns - 9.0%
.40 to .70 microns - 32.0%
Above .70 microns - 59.0%
(See Figure 15 for detailed breakdown).

Douser Speed: The time pulse width at half maximum irradiance is 0.10 seconds to continuous open.

2.2.3 Performance

The maximum irradiance is measured with the Hy-Cal, Asympotic Calorimeter, Model C-1300, with an accuracy of ±3 percent and a repeatability of ±1 percent. The total energy is determined as described in Section 2.2.4 with an accuracy of ±2 percent. The uniformity of irradiance will be ±15 percent or better. The hexagonal shaped beam measures 6.73 centimeters across the flats.

It should be pointed out that the present irradiance capability can be increased at the sacrifice of beam size. A new light pipe could be fabricated that would condense the energy onto a smaller target size at a higher irradiance level. The uniformity of irradiance would also increase.

2.2.4 Sample Calculations

The total incident energy (Q total) for each specimen is determined

FIGURE 15

Table
Thermal Pulse Facility
Spectral Energy Distribution

Band Number	Bandwidth (Microns)	Normalized Data WRT 1.0 Solar Constant (w/m^2)
1	0.25-0.35	69.2
2	0.35-0.40	46.7
3	0.40-0.45	60.3
4	0.45-0.50	75.2
5	0.50-0.60	143.1
6	0.60-0.70	139.7
7	0.70-0.80	128.7
8	0.80-0.90	170.3
9	0.90-1.00	155.0
10	1.00-1.20	117.7
11	1.20-1.40	53.1
12	1.40-1.50	29.0
13	1.50-1.80	49.3
14	1.80-2.20	35.3
15	2.20-2.50	22.5

NOTE: One solar constant equal 1353 w/m^2

2.2.4 Sample Calculations (cont.)

by integrating the thermal pulse curve. This can be easily accomplished by measuring the area under the curve with a planimeter, then establishing a scale factor in order to determine the amount of energy contained within the area in terms of Joules/sq. meter per square inch of paper. This sensitivity is applied to the total measured curve area to obtain the total incident energy. The total incident energy is then divided by the peak irradiance for presentation purposes as Q_{total}/q_{max} in seconds.

2.2.5 Typical Test

The typical thermal nuclear test consists of two pulses, one of very short duration, microseconds, and the second lasting from 1 to 10 seconds. The test schedule is normally for 3 days. This provides one day of test time at each of the pulse requirements. One day is needed in between the two test conditions to re-align and calibrate to the second pulse parameters. A test report is prepared which includes the calibration curves of the Nuclear Therma Pulse Test.

2.3 Photovoltaic Concentrator Facility

The Photovoltaic Concentrator Facility is used to measure the Voltage Current (V-I) characteristics of high solar intensity solar cells. Individual cells or solar cell arrays can be measured to output voltages of 15 VDC and 100 amperes. On a typical 50mm cell, this would require a single XM-300 source module providing an irradiance of 200 solar constants (2.7×10^5 W/M^2). Temperature control of the cell at the high intensity is maintained at 28°C ±1°C. The cell characteristics are obtained with a Compucorp Minicomputer to obtain high speed data acquisition. Data is stored on floppy disc, availab for immediate analysis and printout.

2.4 Special Application

Additional application of the High Flux facility have been in the areas of transpiration cooling of rocket engine nozzle materials under high irradiance (Appendix No. 1), rocket engine fuel tank burn-out when exposed to a hot nozzle, and high power laser material studies.

2.4 Special Application (cont.)

For example, a rocket engine fuel tank insulation material was exposed to a high flux in a vacuum chamber, and the sample weight loss, temperature, and chamber pressure continuously monitored (see Figure 16).

In summary, the Boeing High Flux Facility is considered to be one of the best available. A comparison of different facilities is indicated in Appendix No. 2.

THE BOEING COMPANY

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EQUIPMENT AND CAPABILITIES

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ISSUE NO.	ISSUED TO

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		SHEET NO.	REV LTR	SHEET NO.	REV LTR			SHEET NO.	REV LTR	SHEET NO.	REV LTR
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35	A										
36	A										

REVISIONS			
LTR	DESCRIPTION	DATE	APPROVE
A	Completely revised to include new capabilities at the Kent Space Center, Kent, Washington	5-13-6	
B	Sheet 9 - Changed paragraph 4 to read "A 8-inch-diameter ultra-violet grade quartz window"	5-26-7	
	Changed paragraph 7 to read "Chamber 4 is provided with six $1\frac{1}{2}$-inch passthroughs, three 6-inch passthroughs, and one 8-inch passthrough in the back used for the solar window."	5-26-7	
B	Sheet 25 - Changed photograph to show upgraded Chamber H-4	5-27-7	
A	Sheet 36 - Added Photo of A-7000 Solar Alignment Area	5-27-7	

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SPACE ENVIRONMENT SIMULATION LABORATORY—
EQUIPMENT AND CAPABILITIES

ABSTRACT

The Space Environment Simulation Laboratory is located in Building 18-24 at the Kent Space Center, Kent, Washington. The laboratory consists of two sections: a Materials and Components Laboratory and a Thermal-Vacuum Laboratory. These will be discussed separately since the sizes of chambers and the types of tests carried out in them differ markedly. In addition, a variety of support equipment is used in common by both laboratories and will be discussed separately.

KEY WORDS

chamber
components testing
cryogenics
cryopumping
diffusion pumps
environment simulation
flanges
ion gauges
ion pumps
manlock
materials testing
solar simulation
space testing
subcooling
thermal vacuum
torr
vacuum testing

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1.0 MATERIALS AND COMPONENTS LABORATORY

The purpose of the Materials and Components Laboratory is to perform tests on space vehicle components under thermal-vacuum conditions, with or without solar simulation, and on materials useful for present or future space vehicles, usually with full-spectrum or high ultra-violet solar simulation. Work may be performed as a contract requirement on space vehicles for which Boeing has prime responsibility, on a subcontract basis from other prime contractors, on a Company research basis in connection with contract negotiation, or on a basic research budget for the development of new and better materials or components for use on future Boeing space vehicles.

The laboratory contains nine vacuum systems of widely varying size and complexity, plus a 10-inch diffusion pumping system which is used in conjunction with special chambers fabricated for testing odd-shaped components not readily tested in standard chambers. The following sections describe the dimensions, design, and performance of the individual chambers.

1.1 CHAMBER 1 (BAC 190233)

The working volume of Chamber 1 is 3 feet in diameter by 3 feet deep (27 cubic feet), shielded by a black-painted 100°K shroud, and incorporating helium cryopanels to increase the pumping speed.

This chamber is fabricated entirely of stainless steel and is provided with bakeout heaters to allow bakeout at 400°C if desired. A 12-inch quartz window is available for solar simulation purposes, located in the door on the horizontal centerline 60 inches from the floor.

The rough pumping system consists of a 50-cfm Welch mechanical pump and a 300-cfm Heraeus Roots type mechanical booster, liquid-nitrogen trapped, pumping through a Varian 6-inch right-angle valve. Fine pumping is accomplished by a 2400-liter-per-second ion pump plus titanium sublimation pump using a 3000-square-inch LN_2-cooled substrate, the combination having a total speed of 50,000 liters per second. The angle-fin helium cryopumping array increases the total speed to 100,000 liters per second for air.

The system will pump to 1×10^{-10} torr in 24 hours, with a continuous gas handling capacity for air of 4×10^{-6} torr-liters per second at this pressure. The ultimate pressure (without gas leak) is 1×10^{-11} torr.

In addition to the 12-inch penetration for the solar window, the chamber is provided with four 1½-inch ports, five 6-inch ports, and three 8-inch ports for various types of passthroughs used in tests, including thermocouples, liquid lines, mechanical passthroughs, etc.

1.2 CHAMBER 2 (BAC 528069)

This unit is capable of handling a work package 18 inches in diameter by 18 inches high. It is used primarily for testing of instruments a components for space vehicles and for bearing tests under ultra-high vacuum conditions.

The unit is a vertical one, using a 50,000-liter-per-second ion/titan sublimation pump with an associated liquid-nitrogen shroud and helium cryopump. Roughing is by means of a portable roughing system also us on other systems in the laboratory.

Penetrations include twelve 1½-inch flanges, three 6-inch flanges, an one 12-inch flange (used for a solar window if desired). Metal seals are used for all penetrations, including the 36-inch-diameter main se

Bakeout provisions permit bakeout up to 300°C if required. Readouts for Bayard-Alpert and General Electric penning gauges are provided in the console.

The system will reach 1×10^{-10} torr in 12 hours with a gas load of 4×10^{-6} torr-liters per second, and an ultimate of 5×10^{-12} torr wi out gas leak.

A second base is provided for this chamber, which is arranged for tes ing various types of bearings. The fixturing includes a bearing moun adjustable radial and axial loading systems, torque readout transduce and a vacuum-sealed drive system capable of either rotary or oscillat ing motion. A quartz tube infrared heater is provided which permits bearing operation at temperatures up to 1500°F, as well as at cryogen temperatures.

1.3 CHAMBER 3 (BAC 528068)

This chamber is a substantial duplicate of Chamber 2, except that the spare base has been designed for the acceptance of a Ling 300 Vibrato for vibration testing in vacuum if required.

The pumping units on Chambers 2 and 3 are interchangeable so that eit pump may be used with any base, permitting great flexibility in opera tion and the build-up of tests on one base while the pumping units co tinue operation on the other bases.

Work space, instrumentation, and performance are the same as on Chamber 2.

1.4 CHAMBER 4 (BAC 528067)

This system consists of a stainless steel vessel approximately 30 inc in diameter by 30 inches long. Inside the system there is installed aluminum shroud providing liquid-nitrogen cooling of all surfaces exp

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1.4 (Continued)

to the inside, except where penetrations are required for solar simulation, passthroughs, etc. All surfaces facing the work are painted with a black high-emissivity coating giving an emissivity of approximately 0.95 facing toward the work. External surfaces are left bright in order to minimize heat loss. Shielded behind the projecting fins of the array are helium-cooled cryopumping surfaces which, when cooled with 18°K helium gas, serve as active pumping elements for condensable gases in the chamber.

The system is rough pumped by a portable roughing system consisting of a mechanical pump, blower, and liquid-nitrogen cold trap. Fine pumping is accomplished by an Ultek 1200-liter-per-second ion pump equipped with its own internal bakeout heaters and working in conjunction with a water-cooled titanium evaporator unit. The maximum pumping capacity is approximately 11,000 liters per second.

With all pumping systems operating and the shrouds cold, the system is capable of reaching a pressure of 1×10^{-10} torr in 24 hours with a dry nitrogen inleak of 5×10^{-7} torr-liters per second, or with a hydrogen inleak of 2×10^{-7} torr-liters per second. With no inleakage, the system is capable of reaching a pressure of approximately 2×10^{-12} torr in 24 hours. Actual pressures obtained on a particular test will, of course, depend upon the outgassing rate of the specimens being tested.

A 8-inch-diameter ultra-violet grade quartz window is provided at one end of the chamber, through which a beam of spectrally matched solar radiation can be passed into the chamber and on to the specimens. If a non-collimated beam is used, a diameter of approximately 12 inches can be irradiated at an intensity of one sun (1400 watts/square meter) for continuous tests up to 10,000 hours in length.

The net working space within the chamber, with the shrouds in place and with a sufficient distance between the specimen and the shrouds for proper operation, is 15 inches in diameter by 12 inches long.

The system pressure is monitored on two vacuum gauges. One is a Varian nude Bayard-Alpert gauge, useful down to pressures of approximately 5×10^{-10} torr; the other is a General Electric triggered penning gauge, useful down to pressures of approximately 10^{-13} torr.

Chamber 4 is provided with six 1½-inch passthroughs, three 6-inch passthroughs, and one 8-inch passthrough in the back used for the solar window. All-metal seals are used, including a 30-inch copper wire seal of the Wheeler type on the door.

1.5 CHAMBER 5 (BAC 529788)

This chamber consists of a stainless steel shell provided with an inner liquid-nitrogen cooled shroud and helium cryopumping surfaces similar to those used on the other chambers. The chamber is pumped by means of

1.5 (Continued)

a 10-inch diffusion pump backed by a 4-inch diffusion pump and by a mechanical backing pump with liquid-nitrogen traps provided to preven oil vapors from getting from the pumps and into the chamber. The net pumping speed in operation is sufficient to pump a dry nitrogen inleal of 5×10^{-7} torr-liters per second, at a pressure of 1×10^{-10} torr, in not more than 24 hours including time for bakeout and cooldown of shrouds. The ultimate pressure without gas leak is 5×10^{-12} torr.

The system is provided with a 6-inch-diameter ultra-violet grade quar window for the admission of solar simulation to the specimens.

Chamber 5 has a net working space of 15 inches in diameter by 12 inch long inside the shrouds. The chamber instrumentation includes one Varian Bayard-Alpert type ionization gauge and one General Electric triggered penning gauge for reading chamber pressure.

The chamber is provided with a bakeout system which permits baking th entire chamber to a temperature of 400°C, followed by accelerated coo down if desired.

The chamber is provided with thirteen 1½-inch flanges and three 6-inc flanges. One of the 6-inch flanges, located in the door, may be used as a solar window. All-metal flanges are used, including a gold wire corner seal for the 30-inch door.

1.6 CHAMBER 6 (BAC 528065)

This chamber is similar to Chamber 5, except that it lacks a cryogeni shroud. The net working space is 30 inches in diameter by 36 inches long, and is complete with bakeout system and controls.

The pumping speed is sufficient to permit reaching a pressure of 1×10^{-10} torr in 24 hours with a gas load of 1×10^{-8} torr-liters per second. Ultimate pressure without gas leak is 3×10^{-11} torr. The actual pressure reached during a test will depend on the outgassing rate of the test items. Usual working pressures are from 1 x 10 to 1×10^{-7} torr.

Three 1½-inch-diameter and nine 2-inch-diameter flanges are available for passthroughs, plus a 6-inch flange in the door for use with a sol window when required. Viton O-ring seals are provided on both 30-inc doors.

1.7 CHAMBER 7 (BAC 528064)

This small chamber, having a working space of 12-inches in diameter t 18 inches long without shrouds, is primarily designed for experimenta work in the field of ultra-clean, ultra-high vacuum. However, it is used for some types of ultra-violet materials tests.

1.7 (Continued)

The system is roughed by means of two molecular sieve type sorption roughing pumps and is then pumped during bakeout by means of a 50-liter-per-second ion pump. At the end of the bakeout period, this pump is valved off by means of a bakeable valve; and a 100-liter-per-second ion pump, which has been previously baked, is connected to the system to pump it during the actual run. Under these conditions, pressures as low as 2×10^{-12} torr have been obtained in the empty chamber. Actual pressures during any particular test depend, of course, on the outgassing rate of the test specimen, which must be small to permit use of this chamber since the net pumping speed is only 100 liters per second. If the test specimen is small in dimension, it is also possible to use titanium sublimation to boost the pumping speed by a factor of 5 to 10, depending on the pressure. However, when so used, the upper part of the chamber will be coated by the evaporated titanium; this method is, therefore, useful only when the specimen is not adversely affected by this condition or can be suitably shielded.

1.8 CHAMBER 8 (BAC 534170)

This chamber consists of a stainless steel shell 3 feet in diameter by 6 feet 6 inches long, to which is attached a full-opening ion pump having a pumping speed of 3000 liters per second at 1×10^{-5} torr or lower. A small liquid-nitrogen cooled shroud inside the ion pump provides additional pumping speed for water vapor. The net working space of 3 feet in diameter by 6 feet long provides room for reasonably large space vehicle components or material test setups.

The system is roughed to 1×10^{-4} torr by a portable roughing cart using a mechanical pump, Roots blower, and cold trap. The ion pump is then energized and pumping continued to a pressure determined by the specimen outgassing. The ultimate pressure for the empty chamber is 1×10^{-10} torr.

The control console provides ion pump power supplies, pump bakeout control, and a General Electric penning type vacuum gauge control.

The chamber provides seven 1½-inch flanges, one 4-inch flange, two 6-inch flanges, and one 12-inch flange in addition to a 6-inch door flange for a solar window when required.

Chamber 8 has currently been equipped with additional accessories to permit introduction of electron and proton beams for material studies.

1.9 CHAMBER 9 (BAC 534169)

This chamber is in all external respects a duplicate of Chamber 8.

1.9 (Continued)

Internally, there is provided a liquid-nitrogen cooled shroud with an internal cryopump array cooled by cold, dense helium gas to 14°K. By this means, an internal work area 20 inches in diameter by 48 inch long may be pumped to pressures in the range of 10^{-13} to 10^{-14} torr. Pressure within this extremely high-vacuum area is measured by a quad rupole mass spectrometer and a special extended-range cold cathode gauge of the Redhead type. Uncertainty as to the final low pressure reached is the result of the lack of calibration methods of high accu racy for instrumentation used at these pressures.

Quartz solar windows are provided in both the chamber and the shroud to permit solar simulation to be achieved within the extremely high-vacuum region.

USE FOR TYPEWRITTEN MATERIAL ONLY

2.0 THERMAL-VACUUM LABORATORY

The Thermal-Vacuum Laboratory is located in a high-bay section of the 18-24 Building at the Kent Space Center, Kent, Washington. The three chambers contained therein provide thermal-vacuum testing, with or without solar simulation, of entire space vehicles or large portions thereof. Two of these chambers provide a clear work space of 6 feet in diameter by 8 feet high, and the other provides a work space of 28 feet in diameter by 40 feet high. The handling of parts in the high-bay area is by means of a crane with a 30-ton primary hoist and a 1-ton auxiliary hoist. The hook-to-floor clearance is 85 feet.

2.1 CHAMBER A

This chamber is an elongated spheroid, fabricated of stainless steel, 40 feet in diameter by 50 feet high. It is provided with an inner black-painted shroud 30 feet in diameter by 40 feet high which may be cooled by circulating liquid nitrogen at 110°K or less.

~~Pumping~~ Systems

The chamber is pumped by three types of pumping systems. Roughing is accomplished by a chain blower system having six mechanical pumps with a total speed of 1800 cfm, a second-stage Roots type blower with a speed of 3000 cfm, and a first-stage blower with a speed of 6000 cfm. These pumps are capable of reducing the chamber pressure from atmosphere to 50 microns or below in approximately two hours, at which point cooling of the liquid-nitrogen shrouds may be started.

In approximately two hours, cooldown of the shrouds can be completed, at which time the pressure will normally have fallen to 5 microns or below (5×10^{-3} torr), and the ion pumps may be energized.

The high-vacuum pumping system consists of two identical modules containing a 2-watt helium refrigerator, a 2000-liter-per-second ion pump, and a 100,000-liter-per-second bulk titanium sublimator. The small helium refrigerators have a nominal pumping speed of 1000 liters per second at 1 micron. These are used to lower the chamber pressure sufficiently to permit starting the ion pumps. The total pumping speed for each module at 1×10^{-6} torr and below is approximately 120,000 liters per second.

2.4KW

In approximately two hours after the start of shroud cooldown, the gaseous helium refrigerators may be started and connected to the helium cryopanels. Cooldown of these panels to 20°K or below will require 4.5 to 6 hours, at which time cryopumping will begin. The speed of the cryopanels, for condensable gases (all gases except helium, hydrogen, and neon), is in excess of 3,800,000 liters per second. The pressure resulting from this increase of pumping speed will depend entirely on the gas load being produced by the item under test. For the clean, dry, and empty chamber, pressures of 3×10^{-10} torr have been achieved. With

2.1 (Continued)

actual test articles in the chamber, pressures of 1×10^{-7} torr or lower can usually be achieved after a few hours of pumping, even with fairly large outgassing loads.

For tests not requiring the lowest possible pressures, and with normal outgassing loads, the helium-cryopumping step may be omitted and the chamber operated with ion and sublimation pumping alone (plus liquid-nitrogen cooling of the shrouds if desired), producing pressures in the 10^{-6} torr range, adequate for most thermal-vacuum testing.

The liquid-nitrogen shrouds in the chamber are used to approximate the heat sink of space. They are fabricated of aluminum sheet with internal liquid-nitrogen passageways and are painted with a black absorptive pain having an emissivity of over 0.95. When cooled with circulating, pressurized liquid nitrogen, they have a total heat-absorptive capacity of 300 kilowatts at 140 watts/square foot intensity, while still maintaining a temperature of not over 110°K on their surfaces. A warmup system is provided to return the shrouds to 290°K in six hours.

Manlock

A 10-foot-diameter by 16-foot-long manlock is provided for access to the chamber. The manlock is divided into two sections by a removable, vacuum-tight bulkhead, with provision for independently pumping the two sections. Three 4-foot by 7-foot vacuum-tight doors are provided to permit personnel access from the balcony to the outer lock, from the outer lock to the inner lock, and from the inner lock to the chamber. In addition, the inner door is mounted in a 10-foot-diameter movable gate valve which can be operated under vacuum. When open, the valve provides unobstructed 10-foot-diameter access to the interior of the chamber. The locks can be rough pumped to simulate boost profile.

Chamber Openings and Penetrations

Access to the interior of the chamber is through a full-opening, 30-foot-diameter lid which may be removed by the crane to a storage position at the balcony side of the chamber. A 6-foot-diameter lid is provided in the center of the 30-foot lid, for use for special test purposes.

Penetrations include eight 12-inch-diameter penetrations at the 20-foot level, seven 12-inch-diameter penetrations at floor level, three 12-inch-diameter penetrations in the lid at the 45-foot level, and four 12-inch-diameter penetrations in the chamber near the lid at the 40-foot level.

There are eight 6-inch view ports on the chamber, three 12-inch view ports in the chamber doors of the manlock, five 12-inch view ports in the manlock, and three 6-inch view ports in the lid equipped with 500-watt spotlights for chamber illumination.

2.1 (Continued)

In addition to the above, there are penetrations for vacuum gauges, gaseous-nitrogen bleeds, cryogenic lines, etc., as required.

Lunar Plane Floor

A lunar plane floor 20 feet in diameter (removable) is provided in the bottom of the chamber and is capable of being heated and cooled independently of the chamber from +325°F to -320°F. This floor is capable of supporting a load of 200 pounds/square foot, up to a maximum of 10,000 pounds total load.

Miscellaneous Capabilities

Vibration within the vacuum chamber is possible through the installation of one or more vibrators mounted directly on the stainless steel floor which is coupled directly to a 1.5-million-pound concrete reaction block under the chamber. Up to two 30,000-pound shakers can be used for this purpose if required.

The 30-foot-diameter chamber lid is provided with three hard points, each capable of supporting 6000 pounds of vertical load, spaced on a 12-foot radius.

A removable aluminum-grating internal platform with guard rails is provided at the manlock level to facilitate setup of equipment and access to the chamber interior.

The sources, power systems, and controls for a 40-zone, 60-kilowatt computer-controlled radiant heat system is provided to allow temperature cycling of test items when required. Quartz lamp arrays, installed inside the chamber, are fabricated to conform to the shape of the item to be tested.

A 24-foot-diameter solar simulator for the large chamber is now under construction. The first increment, 10 feet in diameter, will be operational early in 1968.

Firing tests for small rocket systems may be accomplished by mounting the rocket engines in the manlock and utilizing the main chamber as a thermal and vacuum sink during firing.

2.2 CHAMBER B

Chamber B is an intermediate-size space environment simulation chamber with solar radiation simulation capability. The liquid-nitrogen shroud inside the chamber is a vertical cylinder approximately 8 feet in diameter by 18 feet high. The work space is 6 feet in diameter by 8 feet high with solar simulation on, and is 6 feet by 15 feet with solar simulation off.

USE FOR TYPEWRITTEN MATERIAL ONLY

2.2 (Continued)

This chamber loads from the bottom with the bottom section capable of being removed and dollied so that the test setup may be built up on the floor, remote from the chamber. Material handling equipment available for use with this chamber consists of 30-ton and 1-ton bridge cranes 85 feet to hook, and a hydraulic elevator hoist installed beneath the chamber and used to remove and remate the chamber bottom loading section. This hoist is capable of lifting a 6000-pound net load mounted on the chamber bottom section.

Pumping System

Chamber B is capable of a working pressure of 1×10^{-9} torr with a 1×10^{-5} torr-liter-per-second nitrogen gas load. The "dry" chamber ultimate pressure is below 3×10^{-11} torr. Pumpdown to working pressure takes less than 24 hours under full gas leakage load. Pumpdown is accomplished by a 1250-cfm chain blower and/or the Chamber A roughing system, a 110,000-liter-per-second baffled ion and sublimation pump, and a 20°K helium cryopump capable of 500,000 liters per second using a "Santeler" pumping surface array. This type of fine pumping system allows this chamber to be used for tests which require ultra-clean conditions.

The shrouds used to approximate the heat sink of black space are constructed of black-painted aluminum and cooled with circulating liquid nitrogen. The shrouds for Chamber B have a 100-kilowatt total heat load capability at 280 watts/square foot intensity without exceeding 110°K at their surfaces.

Solar Simulator

The solar simulator built into this chamber provides a collimated beam 42 inches in diameter by 8 feet high. The light is generated by a bank of nineteen 2.5-kilowatt xenon short-arc lamps located in a lamp chamber adjacent to the vacuum chamber. Each lamp is surrounded by an aconic collector which directs the beam by means of a folding mirror through a diagonal connection into the vacuum chamber through an optical train consisting of quartz condensing elements and projection elements. Uniformity filters and spectral filters are provided in the optical system to shape the beam both in uniformity and spectrally to that of the sun outside the atmosphere.

Inside the vacuum chamber, a 5-foot parabolic collimating mirror of off-axis orientation projects the collimated beam downward into the working area. The resulting collimation half angle within the working area is 1.45 degrees. Beam uniformity, measured by a 4 cm by 4 cm calibrated solar cell, is ±5.0 per cent in any one plane or ±8.0 per cent over the entire 42-inch by 8-foot working volume. Beam intensity is variable between 400 and 1720 watts/square meter. Spectral match has been carefully measured over the entire range from 0.26 to 2.60 microns in 16 bands; the greatest deviation from solar spectral distribution

2.2 (Continued)

(-49.6%) occurs in the band from 0.26 to 0.33 microns. Over the remainder of the range, the deviations are from +29.2% to -13.6%, with the total integrated value being within ±5.0% of the space sun values as given by the Johnson curve.

The control for the entire system is from a console located in the Control Room. From this point, performance of each individual lamp can be metered, as well as the integrated total energy flux of all the lamps. With this system, each individual lamp views the entire irradiated area. Thus, failure of any individual lamp affects neither the uniformity nor the spectral match, but only the total intensity which can generally be compensated for by increasing the output of the remaining lamps, thus enabling the continuation of the test in progress.

Penetrations

The chamber bottom section has four 1½-inch ports, six 4-inch ports, and six 12-inch ports for test instrumentation. The chamber itself has four 1½-inch ports, two 4-inch ports, and four 12-inch ports. Four windows are also available for viewing the inside of the chamber.

Bakeout System

A separate bakeout heater of the gas-fired, air-circulation type is available, with suitable controls, duct work and vents, for use interchangeably with Chamber C. This permits bakeout of the system at temperatures of up to 300°C with the solar mirror removed and the entrance window for the solar simulator blanked off. This is normally used only when reconditioning the chamber after major contamination has occurred. A low temperature (200°F maximum) can be employed to hasten shroud warmup when desired, using the heater at minimum setting.

2.3 CHAMBER C

This chamber is a substantial duplicate of Chamber B, with the solar simulation omitted. In consequence, its maximum loading dimension is 6 feet in diameter by 8 feet high. The vacuum capabilities, pumping speed, shroud arrangement, and passthrough penetrations are identical to Chamber B.

3.0 ACCESSORY EQUIPMENT

3.1 CRANE

To permit charging of large test articles into Chamber A, as well as handling of the chamber lid and general handling capability within th high-bay area, there has been provided a large traveling bridge crane The main hoist has a 30-ton capacity and a clearance height to the ho in raised position of 85 feet. A 1-ton auxiliary hoist is also provided. Control of the crane is by means of a control pendant which may be raised, lowered, or traversed along the bridge independent of trolley position. An auxiliary key-locked stationary control positio is provided on the balcony to permit shifting of the control pendant to elevated locations when desired.

3.2 LIQUID-NITROGEN SUBCOOLER

To supply cooling to the various shrouds in both laboratories, a 300-kilowatt liquid-nitrogen subcooler is provided. In this system, liqu nitrogen supplied from two 14,000-gallon-capacity vacuum-jacketed sto age tanks on the building roof is pressurized to approximately 80 psi and pumped through vacuum-jacketed recirculating lines to the various chambers. After circulating through the shrouds, the liquid returns to the subcooler at a pressure of approximately 40 psi. Since the liquid is always pressurized, it does not boil within the shroud passages, but absorbs heat through temperature rise of the liquid from a starting temperature of approximately 80°K to a maximum return temper ture of 110°K. Upon return to the subcooler, the warm liquid is pass through a heat exchanger immersed in liquid nitrogen boiling at atmos pheric pressure, thus again cooling the circulating liquid which then passes to the pumps for recirculation.

The subcooler's maximum output is achieved with a circulation rate of 300 gallons/minute. To achieve this, four 100-gpm nitrogen pumps are provided (1 is a spare) plus one 50-gpm pump which is used alone at times of low demand.

3.3 HELIUM REFRIGERATORS

To provide cryopumping capability, two 1200-watt helium refrigerators are provided. These systems compress helium gas to approximately 300 psi, pass it through a series of heat exchangers, then expand it thro an expansion engine. After cooldown, the gas leaving the expansion engine will have a temperature below 15°K and a pressure of 30 psi. I then passes through vacuum-jacketed, super-insulated lines to the var ious chambers, through cryopump arrays, and back to the refrigerators at a temperature of not more than 20°K for recirculation. The actual cryopump arrays in the chambers must be maintained at a temperature o 20°K or below to permit cryopumping in the ultra-high vacuum range.

USE FOR TYPEWRITTEN MATERIAL ONLY

3.4 CONTROL PANEL

Control of all motors, pumps, and valves for Chambers A, B, and C, plus control of the liquid-nitrogen subcooler, the two helium refrigerators, and the solar simulator in Chamber B, is from a centralized Control Room located on the manlock level on the second floor. An annunciator panel provides visual and audible signals to notify the operator of a variety of equipment malfunctions. An intercom system provides head-set communication over any one of eight channels from the Control Room to all parts of the laboratory, as well as communication between any two or more station locations. Paging systems allow paging either from the control room or from the entrance lobby. One, or at most two, duty operators in the control room can operate all the high-bay chambers plus the nitrogen subcooler and refrigerators.

3.5 500-CHANNEL CSC DATA SYSTEM

Also located in the Control Room is a data acquisition, data processing, and data control system. The system is computer controlled and will service all of the systems in the laboratory simultaneously.

The data acquisition capabilities of the system include: 500 signal conditioning modules which convert voltage, current, or resistive transducer outputs to millivolts; 500 low-speed data channels which are read by the computer at a maximum rate of 200 readings per second; and 40 high-speed data channels with a maximum input rate of 12,500 readings per second. The system has four full-scale input ranges: 10 mv, 20 mv, 50 mv, and 100 mv. Overall system accuracy is better than 0.5 per cent of full scale. The input of each channel is selected by the computer under control of its stored program.

The computer will convert the data from millivolts to physical units (pressure, load, temperature, etc.), compare the data against maximum-minimum and/or rate-of-change limits, record the data on one or two magnetic tape recorders, and/or print selected data on the line printer. Execution can be radically varied from one test to another to meet a wide variety of test requirements. The computer will accomplish high-speed closed-loop and outputting continuously variable control signals through one or more of the forty digital-to-analog converter channels. Up to 40 channels of on-off control are also available.

A master event controller enables the system to handle tests from several chambers simultaneously. Each test is divided into one or more jobs and each job is assigned a priority number and a computer service rate. The service rate can vary from 4000 times per second to once every several hours. The system handles up to 24 jobs simultaneously. While the system is operating, time periods will be available for additional work. These time periods will be used to perform further analysis on recorded data, to prepare document quality test reports, to set up new tests, or to reformat data for additional analysis on the IBM 7094 or Univac 1107 computers at the Boeing Computer Center in Seattle.

3.5 (Continued)

In addition to the system described above, there is also available in the Control Room a simple digital data logging system. This system will handle 200 low-level data points, recording either on magnetic tape (at the rate of 10 points per second) or on "grocery tape" print-out. This system may be used for tests requiring no control function. Data may be reduced on the SDS 910 Computer in the 500-channel data system, another SDS 930 Computer in the same building, or the Boeing Computer Center.

3.6 SOLAR SIMULATORS

Two 2.5-kilowatt xenon short-arc lamp-powered simulator units are available for tests in the Materials and Components Laboratory requiring solar simulation. Lens systems are available to project a collimated beam 6 inches in diameter or a diverging projected beam 13 inches in diameter at one solar constant (1400 watts/square meter) to test specimens inside vacuum chambers. Optically flat, bakeable quartz or sapphire windows are used to transmit the beam into the vacuum chambers. Windows of 4 inches, $5\frac{1}{4}$ inches, 6 inches, and 12 inches are available for use as needed.

In addition, one carbon arc solar simulator is available. This will irradiate a circle 19 inches in diameter using projection optics, or 12 inches in diameter using a collimated beam, to one solar constant.

3.7 MASS SPECTROMETERS (RESIDUAL GAS ANALYZERS)

Two mass spectrometers are available for analyzing residual gases inside vacuum systems during tests. The Consolidated Electrodynamics cycloidal type residual gas analyzer has analyzing capabilities to mass 60 at minimum chamber total pressures of approximately 5×10^{-9} torr. The General Electric sector type instrument has approximately the same mass range but greater sensitivity, enabling it to be used at total pressures down to 1×10^{-10} torr or slightly lower. A new Varian quadrupole mass spectrometer will be available after June 1966.

3.8 ION GAUGE CALIBRATION SYSTEM

A Varian ion gauge calibration unit is used for pre-calibration of all ion gauges used in test work. Calibration is carried out in accordance with the A.S.T.M. Ion Gauge Calibration Standard down to a pressure of 5×10^{-8} torr.

3.9 OTHER EQUIPMENT

An assortment of other equipment is available in the laboratory to support test requirements. Included are four helium mass spectrometer leak detectors, portable pumping units for roughing small ion-pumped chambers, portable pressure gauges, total radiometers, a spectroradiometer for checking solar simulators, and an assortment of electrical meters of a wide variety of types. These are portable devices which may be used as needed in any of the chambers in the laboratory.

APPENDIX E

WESTINGHOUSE ELECTRIC CORPORATION RESPONSE

Westinghouse Electric Corporation

Power Systems Company

Fusion Power Systems Department

Box 10864
Pittsburgh Pennsylvania 15236
(412) 892 5600

FPSP:81-389

September 24, 1981

Dr. P. Y. Hsu
EG&G Idaho, Inc.
P. O. Box 1625
Idaho Falls, ID 83415

SUBJECT: Response to Request for Information on Candidate Westinghouse Test Facility for TPE-II

Dear Peter:

Dr. Rose has asked me to respond to your September 11, 1981 request for information on Westinghouse test facilities that could be considered for use in the TPE-II program. I am enclosing herewith descriptions of several Westinghouse test facilities that we feel have potential applications for the TPE-II program. They include the following:

- Pressurized water coolant loops and test facilities with water exit temperatures up to 300^{o}C and coolant pressures up to 2000 psi.
- Liquid lithium loops (LLP and ELF).
- High surface heat flux test facilities (ESURF, ASURF and plasma arc heating systems).
- Test loops to study forced convective boiling heat transfer to water.

The enclosed material should provide most of the answers to your questions. With the exception of ASURF, all are existing test facilities that can be used in support of TPE-II, provided that 6-9 months of advance notice is given to allow time for scheduling. ASURF is scheduled to be operational in March 1982.

We do not have a test facility that can provide nuclear heating of the type necessary to simulate fusion blanket/shield bulk heating. However, electrical resistance heating to simulate nuclear heat generation is possible with any of the listed test facilities. We have past experience in simulating nonuniform heating with such a facility by spatial variation of electrical resistance of the test piece. The necessary power supplies for electrical resistance heating are generally available.

Further review and discussion will be required to provide a detailed response to your questionnaire concerning the cost of using the test facilities. Obviously, the cost will depend on the type and the scale of the tests required. Once this is better defined, we can supply you with the cost information. Fully qualified and highly experienced professional personnel are available to support these test facilities. This support also includes machine shops and metallographic and chemical laboratories, etc. as described in the enclosed material.

Please feel free to give me (412-892-5600, ext. 5486) or Dr. Rose (412-892-5600, ext. 5422) a call should you need any additional information or have further questions.

Sincerely yours,

J. W. H. Chi

/jm
enclosure

cc: R. P. Rose
T. C. Varljen

PWRSD Test Engineering Laboratory

Hydraulic Model Testing

Miscellaneous hydraulic tests on mock-ups of reactor system parts and components are routinely performed at the Test Engineering Laboratory. Typical of this type of testing are the two discussed below, which were recently completed:

Emergency Core Cooling Flow Distribution: As shown above, a 10 x 10 rod bundle was installed in a plastic housing with a water supply at the top. A grid-collection unit at the bottom of the bundle collected the water as it flowed through the model and diverted it to the measuring tubes at the base. Knowledge of the flow distribution in the bundle was obtained in this manner.

Sample System Mixing Test: This test used one thermocouple to measure the temperature of water from four locations in a reactor. The purpose of the procedure was to determine whether the indication from the single thermocouple was representative of the average temperature of the four water supplies. A mock-up of the mixing chamber was constructed so that hot or cold water—at closely controlled pressure—could be supplied to any of the four inlets. By running combinations of hot and cold inlets and making simultaneous recordings of the various temperatures, highly useful information was obtained.

PWRSD Test Engineering Laboratory

Mechanical Component and Vibration Tests

Full-scale mechanical and vibration tests are performed at the Test Engineering Laboratory on plant and reactor components to prove the reliability of equipment design.

Typical of these tests is the Rapid Refueling Roto-Lok Stud, a stud bolt with breechlock thread. The primary purpose of this testing was to determine the stresses in thread segments for different stud tensions. The Roto-Lok was cyclically loaded to 3,750,000 pounds, and stresses were measured with 140 strategically located strain gages. Data obtained were then converted to stress versus tension.

Another test was a cycling endurance test of a full-size Cable Tray, which is part of the Rapid Refueling System. The purpose of this test was to determine tray loads, deflections, and cable wear during the life of an installation—and 25% beyond. The Cable Tray was positioned on rollers and with one end inserted into a deep pit, the Tray (with cables installed) was raised vertically at one end with a mobile crane to a point 20 feet above ground. With cables hanging over the end of the tray at a 90° bend, the elevated end was moved about 35 feet horizontally, and the opposite end dipped into the pit. This was followed by a return to the starting position for a total of 220 cycles, each one involving alternate 90° bends and straightening of the fully installed cables.

Vibration testing of reactor components is also performed in this Laboratory, using electronically excited shaker heads. Three sizes are available (2 lbs., 50 lbs., and 150 lbs.) for regular scale model testing for frequencies from 5 H_z to 50 H_z.

PWRSD Test Engineering Laboratory

"A" & "B" Loops. Low-Flow/High-Pressure Hydraulic Facilities

These loops are small, high-pressure, stainless steel facilities, used for testing small components and individual parts of larger components under normal working conditions. A canned motor pump circulates water in both "A" Loop and "B" Loop at 150 gpm. Operating temperatures are obtained from the conversion of the pumping power into heat, as well as from external heaters. Typical tests run in these loops are: a. full-scale gate and check valves; b. material corrosion-erosion, with variable water chemistry; and. c. corrosion product release and transport properties of crud.

Characteristics of "A" & "B" Loops

Maximum Flow Rate .... 150 gpm at 300 ft.
Maximum Pump Head .... 335 ft. at 60 gpm
Maximum Allowable Pressure .... 2400 psi
Maximum Allowable Temperature .... 650°F
Normal Working Pressure .. 2000 psi
Normal Working Temperature .... 600°F

PWRSD Test Engineering Laboratory

"D" Loop. Medium-Flow/High-Pressure Hydraulic Facility

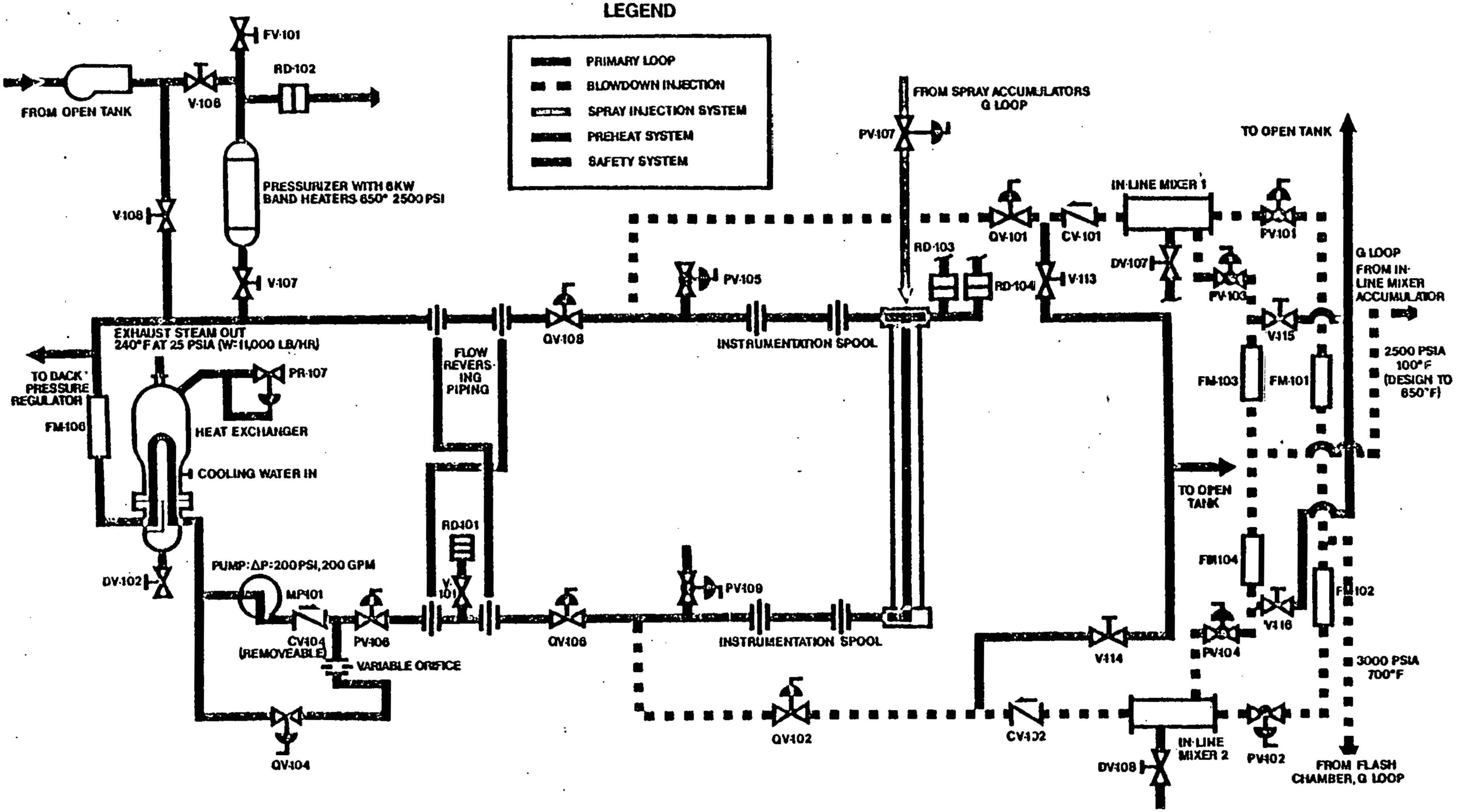
The "D" Loop is a flexible test facility used for demonstrating the interplay of reactor subsystems and evaluating component design concepts. It contains a canned motor pump, which produces a 290 ft head at 3000 gpm. All piping (10-inch Schedule 160) in contact with the primary water is stainless steel. Loop pressure is established and maintained by an air driven charging pump operating in conjunction with a gas loaded back pressure valve. Most of the power required to establish and maintain loop temperature is derived from the circulating pump operation, and 75 Kw of heat is available from electric strip heaters.

The "D" Loop services a 25" ID x 38' long test vessel, which accommodates full-scale models of large PWR core components for operational studies.

Characteristics of "D" Loop

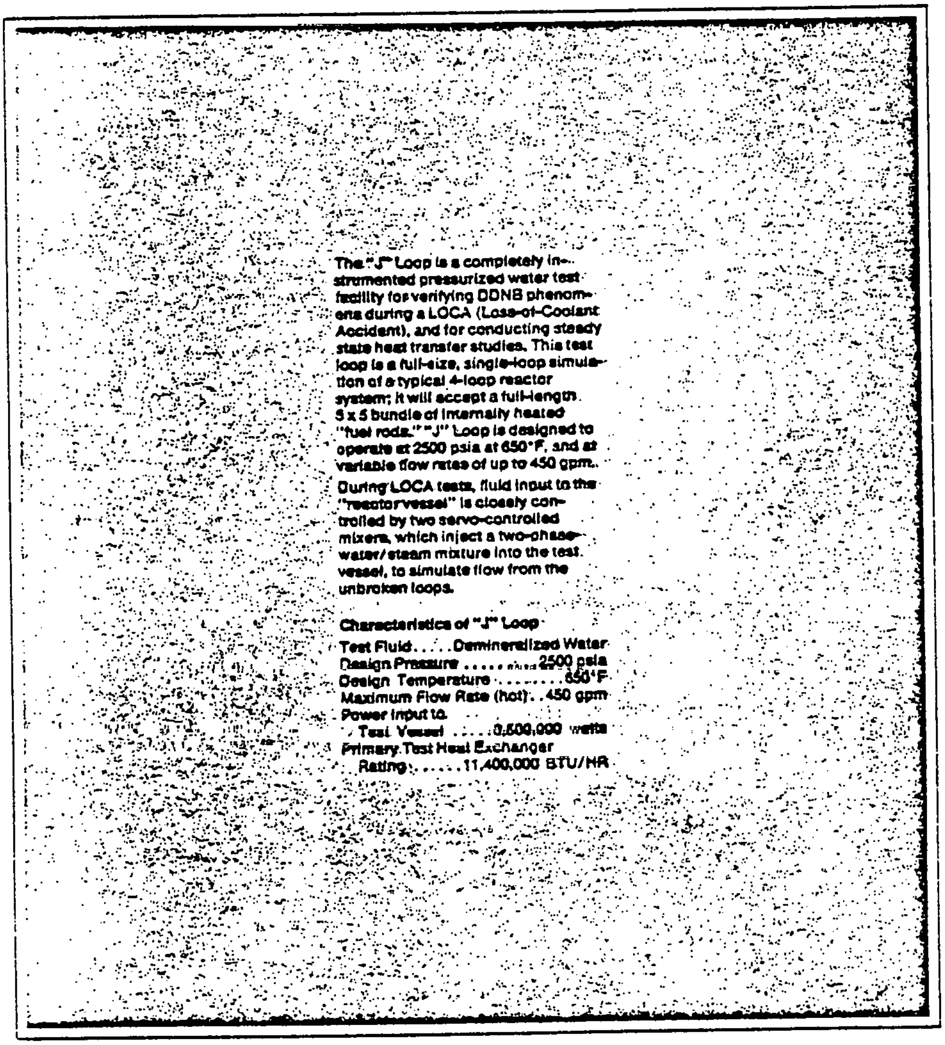
Maximum Flow Rate ..... 4400 gpm
Maximum Allowable Pressure .... 2400 psi
Maximum Allowable Temperature .... 650°F
Normal Working Pressure .. 2000 psi
Normal Working Temperature . 600°F
Pump Head at 3000 gpm ..... 290 ft
Maximum Pump Head .... 340 ft (at 1500 gpm)
Main Loop Flow Measurement .... 10" Venturi
Auxiliary Flow Measurement .... 5" Venturis (2" Branch Lines)

"J" LOOP DDNB TEST FACILITY



PWRSD Test Engineering Laboratory

"J" Loop. Delayed Departure from Nucleate Boiling Heat Transfer Facility



The "J" Loop is a completely instrumented pressurized water test facility for verifying DDNB phenomena during a LOCA (Loss-of-Coolant Accident), and for conducting steady state heat transfer studies. This test loop is a full-size, single-loop simulation of a typical 4-loop reactor system; it will accept a full-length 5 x 5 bundle of internally heated "fuel rods." "J" Loop is designed to operate at 2500 psia at 650°F, and at variable flow rates of up to 450 gpm.

During LOCA tests, fluid input to the "reactor vessel" is closely controlled by two servo-controlled mixers, which inject a two-phase water/steam mixture into the test vessel, to simulate flow from the unbroken loops.

Characteristics of "J" Loop

Test Fluid . . . . . Demineralized Water
Design Pressure . . . . 2500 psia
Design Temperature . . . . 650°F
Maximum Flow Rate (hot) . . 450 gpm
Power Input to
Test Vessel . . . . . 3,600,000 watts
Primary Test Heat Exchanger
Rating . . . . . . 11,400,000 BTU/HR

PWRSD Test Engineering Laboratory

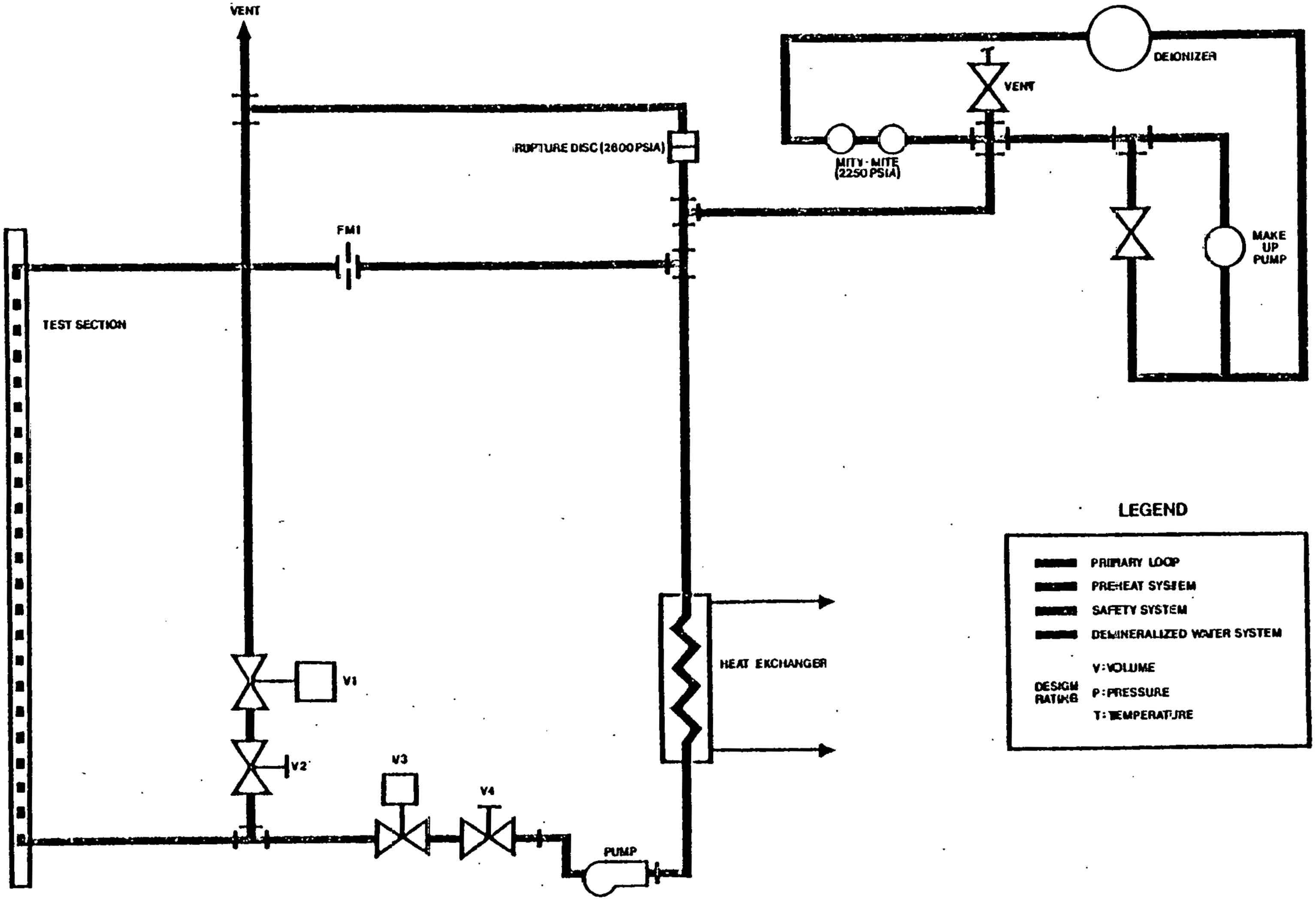
Single-Rod Loop. Heater Rod Development Facility

The single-rod test loop is used to evaluate prototype heater rods and for in-depth study of existing rods in pressurized water systems. The test section of the loop is easily replaced to facilitate the installation of various length and diameter heater rods. The Single-Rod Loop is electrically controlled and operated by one person. Steady state and blowdown at various conditions can be simulated in the loop. The main test section can be replaced with a quartz tube, and DNB phenomenon can be observed on a single rod with a remotely operated camera.

Characteristics of Single-Rod Test Loop

Maximum Operating Pressure	2250 psi
Maximum Operating Temperature	650°F
Maximum Flow Rate	10 gpm
System Capacity	5 gal
Maximum Power Available	200 Kw
Piping Size	1" and 3"

SINGLE ROD HEATER DEVELOPMENT FACILITY



STEAM GENERATOR TEST FACILITY (SGTF)

In 1967, ARD initiated facility design efforts toward a Steam Generator Test Facility (SGTF) for testing of LMFBR sodium-water steam generator models. The facility was commissioned in 1968. The facility consisted of (1) the General Purpose Loop No. 1 (GPL-1) as a sodium source, (2) the Steam-Water Loop (SWL-2) as a source of high-temperature, high-pressure feedwater, and (3) SILO-I, which is a silo-type test enclosure.

The SGTF has been fully-utilized since commissioning on the following programs:

- Aluminum-Bonded Steam Generator Test Program
- Steam Generator Tube-to-Tube Sheet Weld Test Program
- 1MW, Three-Tube Steam Generator Test Program
- LMFBR, Double Wall Full-Scale J-model Steam Generator Test Program
- LMFBR, Double Wall Steam Generator Model Test Program

The facility is currently being modified to conduct LMFBR steam generator model tests in support of the ANL/ Ⓦ Tampa double-wall steam generator development program.

In completing the above programs, ARD has developed a trained staff of engineering personnel, operating crews, and supporting technologies. This staff, which will be applied to the forthcoming Double-Wall Steam Generator Model Program, has been utilized on a 24-hour per day, seven-day per week basis during steam generator testing.

The facility modifications currently underway consist in-part of up-grading the thermal capacity from 1.0 MWt to 2.0 MWt. This up-grading necessitates complete removal of the original SWL-1 Steam-Water Loop, and replacement with an up-graded 2.0 MWt System (SWL-2). This feedwater system up-grading (cost $700K) is part of the overall facility up-grading which included increasing the height of the test silo to 85 feet to accommodate large steam generator models. In addition to up-grading of the GPL-1 sodium supply system to the combined GPL-1/GPL-1A system as a part of the LMFBR Thermal Striping Program, a Distributed Data Acquisition

OPERATING AND DESIGN PARAMETERS

Facility Operating Mode		Recirculation or once-through
Thermal Capacity		2.0 MWt Heat Input (GPL-1/GPL-1A) 2.0 MWt Heat Removal (SWL-2)
Maximum Operating Pressures		330 psig Sodium 2500 psig Water/Steam
Maximum Operating Temperatures		1200°F Sodium 950°F Steam
Maximum Flowrates		400 GPM Sodium (GPL-1/GPL-1A) 30,000 lb/hr Water (Recirculation Flow) 10,000 lb/hr Water (Feedwater Flow)
Transient Capability		To be determined
Third Fluid Service System		Available
Low Flow Capability		Control and measuring capability is available.
Test Enclosure Size		12 feet diameter × 35 feet high
Water Chemistry Capability		CRBRP
	Recirculation Pump	**Feedwater Pump**
Type	2-Stage, Centrifugal Canned Rotor	2-Stage, Turbine Canned Rotor
Rated Flow	30,000 lb/hr (60 gpm)	16,000 lb/hr (30 gpm)
Head & Power	530 psig and 85 kW	320 psig and 42 kW

System has been installed for data acquisition and analysis.

As a part of the continuing Westinghouse corporate support of DOE programs, the five-month test program to be carried-out during FY-1980 (and the FY-1981 subsequent post-test examination effort) is being funded by Westinghouse.

In summary, Westinghouse has established an operational LMFBR Steam Generator Model Test Facility and, in the course of over eleven years of operation (and in excess of 20,000 hours of steaming at test conditions), has developed a trained and competent crew of personnel for the installation, operation, and examination of steam generator models. The combination of GPL-1/GPL-1A, SWL-2, and the related silo test enclosure provides a test bed for the evaluation of full-scale LMFBR steam generator models at prototypic conditions. In addition, the two loop feature of GPL-1/GPL-1A provides the unique capability of thermal transient and shock testing of steam generator models.

PWRSD Test Engineering Laboratory

"G" Loop. Emergency Core Cooling System Facility

The "G" Loop at Forest Hills is a high-pressure, emergency core cooling (ECCS) test facility designed and fabricated to ASME Section I for 2000 psi and 650°F. It consists of a main test section and vessel, exhaust system, piping, separators and muffler, flash chamber steam supply system, and high pressure/low-pressure cooling systems.

This loop is basically designed to obtain test data for analysis of LOCA, for breaks up to and including double-ended pipe breaks for Pressurized Water Reactors. Tests are initiated at simulated conditions existing 8 seconds after the start of a LOCA (Loss-of-Coolant Accident). A typical run consists of constant power and pressure, followed by pressure blowdown, power decay and ECCS.

"G" Loop is capable of performing the following methods of ECCS: *Current, Upper Head Injection (UHI) UHI w/Current* and other core spray systems. It may also be used for constant temperature/pressure small leg break tests (core uncovering tests). These consist of boiling off water at a constant bundle power input until the rods can no longer be cooled.

The "G" Loop test bundle consists of 480 electrically heated rods, 16 grid support thimbles, and 33 spray thimbles bounded by an octagonal stainless steel baffle and arranged as per a 4-loop 15 x 15 rod bundle configuration. The loop is controlled (fully automated during transients) through a PDP-II-DEC-16K computer with a 600 point Computer Products A-D Converter operating at a sweep rate of 40,000 pts/second for data acquisition.

"G" LOOP SYSTEM COMPONENTS & CHARACTERISTICS

COMPONENT	MATERIAL	RATED PRESSURE & TEMP.	TYPICAL OPERATING P&T
Test Vessel	Carbon Steel	2000 psi @ 650°F	1000 psi @ 545°F
Downcomer Side Tank	Carbon Steel	2000 psi @ 650 F	1000 psi @ 545°F
In-Line Mixer	Carbon Steel	2000 psi @ 650 F	1000 psi @ 545 F
Mixer Accumulator	Stainless Steel	2500 psi @ 650°F	1800 psi @ 100 F
Flash Chamber	Carbon Steel	3000 psi @ 700°F	2800 psi @ 660 F
Separators Nos. 1 & 2	Carbon Steel	2000 psi @ 650°F	1000 psi @ 545 F
Spray Accumulators Nos. 1 & 2	Carbon Steel	2000 psi @ 550 F	1800 psi @ 150 F
Spray Accumulator No. 3	Stainless Steel	2500 psi @ 850 F	1800 psi @ 150 F
Reflood Tank	Stainless Steel	Atmos @ 2 2°F	Atmos @ 150 F
Primary Piping	Carbon Steel	2000 psi @ 850°F	1000 psi @ 545°F

PWRSD Test Engineering Laboratory

"E" Loop. Low-Flow/Low-Pressure Hydraulic Facility

The "E" Loop is a low-pressure, six-inch, stainless steel loop, with two circulating pumps. These pumps may be connected in parallel, giving 2000 gpm at 130 ft. head, or in series, giving 1000 gpm at 260 ft. head. Flow and vibration studies are conducted with this loop, and, because of its low pressure, plastic models for visual observation or photography may be used. In addition, a 4-inch Rockwell water meter in a branch line permits the calibration of flow meters up to 800 gpm.

Characteristics of "E" Loop

Maximum Flow
Rate .... 2000 gpm at 130 ft.
1000 gpm at 260 ft.

Maximum Working
Pressure .... Pump Head

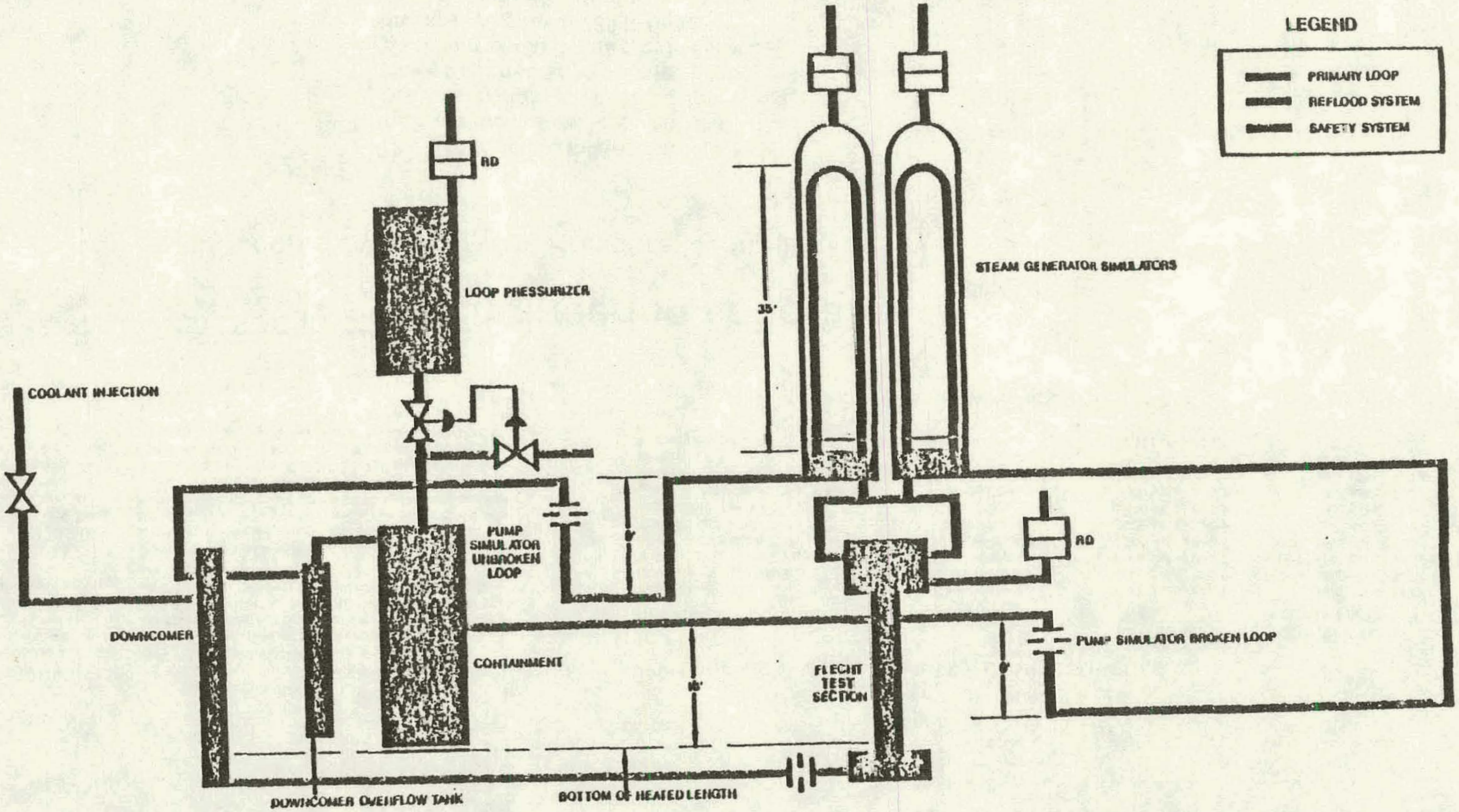
The FLECHT-SET is a low pressure facility, designed to provide experimental data on the influence of system effects on ECCS during the reflood phase of a Loss-of Coolant Accident (LOCA).

The facility consists of a once-through system, including an electrically heated test section ("fuel rods" and housing), accumulator, steam generator simulators, pressurizer, catch vessels, instrumentation, and the necessary piping to simulate the reactor primary coolant loop. Data acquisition is accomplished through a PDP-II DEC-16K Computer with a 256 point Computer Products A-D Converter, operating at a sweep rate of 1200 pts/second.

Characteristics of FLECHT-SET

100 Rod Bundle
Maximum Power . . . . 1000 Kw
Maximum Bundle
Flooding Rate . . . . 86 gpm
Water Temperature
Range . . . . 100-200 F
System Pressure . . . . 0-60 psia

FLECHT-SET PHASE B TEST FACILITY



PWRSD Test Engineering Laboratory

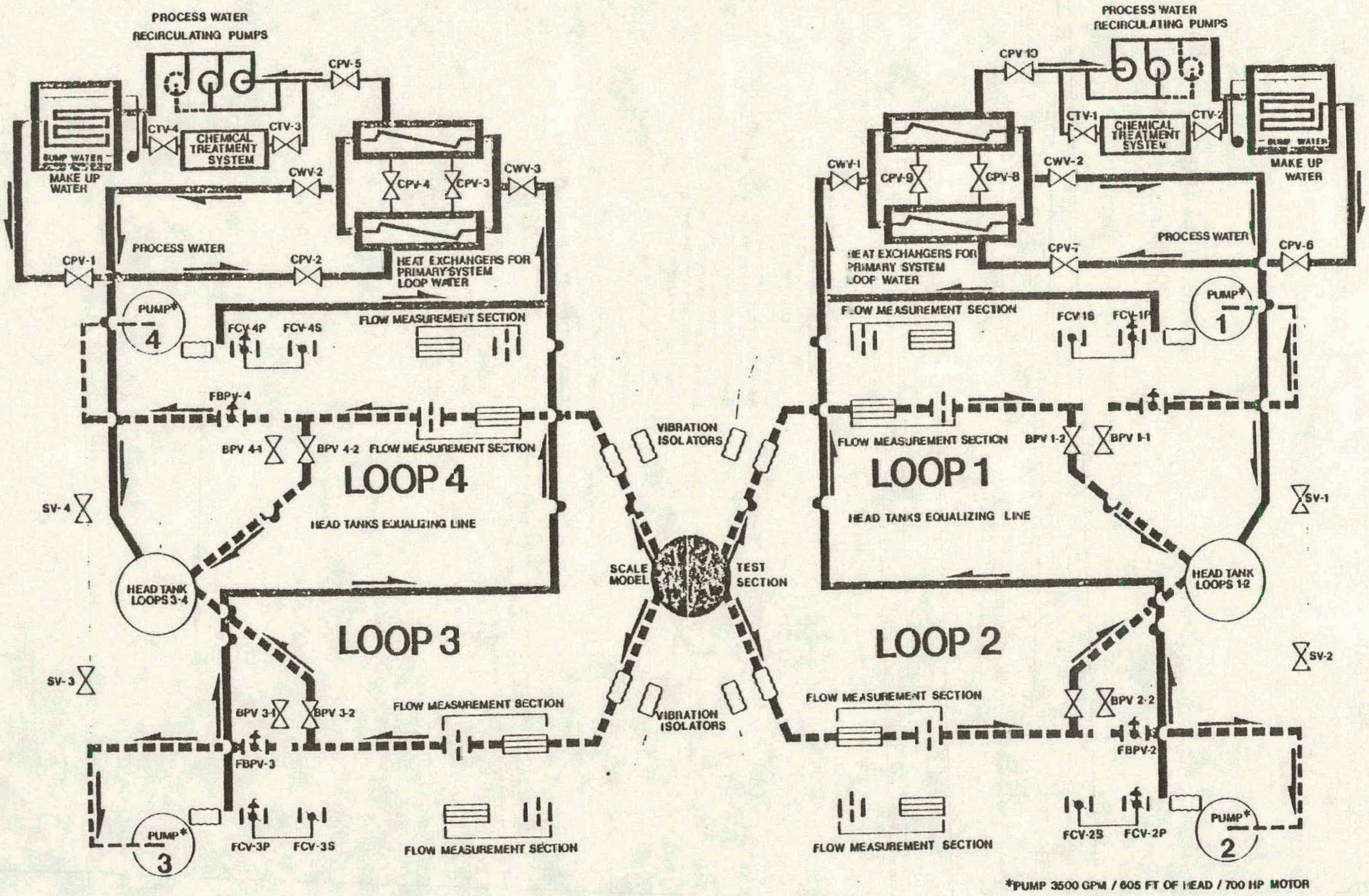
"H" Loop. High-Flow Hydraulic Facility

The "H" Loop constitutes a versatile hydraulic facility, capable of supplying 14,000 gpm of water at a developed head of 600 ft. and at temperatures as high as 200°F. This 4-loop system can simultaneously handle either full-scale prototype test assemblies, or one large-scale reactor model. The major purpose of "H" Loop is to permit the use of 1/7 scale reactor models and full-scale fuel assemblies for conducting mixing studies, flow distribution studies, and similar low-temperature/low-pressure hydraulic tests.

Characteristics of "H" Loop

Maximum Flow Rate . . . 14,000 gpm
Pressure Drop Across Vessel Model . . . . 120 psi
Minimum Vessel Outlet Pressure . . . . 10 psig
Flow Accuracy . . . . ½ %
Water Temperature Range . . . . 70-200°F
Maximum Loop-to-Loop Temperature Variation . . . . . . 2°F
Maximum Loop-to-Loop Flow Rate Variation 3%

"H" LOOP HIGH FLOW-HYDRAULIC FACILITY



GENERAL PURPOSE SODIUM LOOPS (GPL)

The General Purpose Loops No. 1 and No. 2 (GPL-1, GPL-2) were commissioned in 1967 and 1971, respectively. These high-temperature sodium test facilities were designed, as their names imply, as general loops for utilization in a wide range of concurrent test programs. This approach deviates from the "one facility - one program" concept, resulting in fuller utilization of both facilities and manpower on LMFBR Programs.

Both GPL's have employed the "parallel test section" approach over their operational life. Facility design is such that test articles can be installed, removed, or serviced in these parallel test section areas while the facility continues operation on other on-going programs. The result is full facility utilization with optimum economic benefits to the customer. Facility operating costs to be incurred by any one program can be minimized by (1) sharing operating costs with concurrent programs, or (2) operating with more than one test section.

In both GPL's, the basic facility has been expanded to include parallel sodium test circuits, resulting in increased versatility. The results have been to increase the versatility of the facility. The GPL-1 facility is now a two-loop system, and GPL-2 is three-loop. The respective facility expansions identified above provide facility versatility by (1) providing increased flow and power capability for future test programs, and (2) permitting parallel, concurrent operation of the various systems at different operating conditions. An additional capability that the multiple-loop feature brings to the GPL-1 and GPL-2 facilities is the ability to impose thermal transients and thermal shocks on test articles through the use of circulating sub-loops at different sodium temperatures.

Because of the inherent time-consuming efforts associated with facility start-up and shutdowns, both facilities are operated on a 24-hour day, continuous basis. Actual continuous operating periods for the two facilities have ranged from one week to six months.

Both facilities are currently in use on the following programs:

Facility	Program
GPL-1	Radial Blanket Heat Transfer Test Program
GPL-1	LMFBR Thermal Striping Evaluation Program
GPL-1	LMFBR Double-Wall Steam Generator Test Program
GPL-1	CRBRP Dynamic Friction (Seismic) Test Program
GPL-2	CRBRP Primary Shutdown Systems Program

In summary, ARD currently has two operational Sodium Test Facilities with capability for multiple, concurrent test program implementation. The upgrading and expansion over the years, including the recent addition of Distributed Data Acquisition Systems at both facilities, provide versatile "test beds" for future DOE programs. The application of the parallel test section concept permits maximum economic benefits to the customer; and the application of the "multiple-loop" concept provides facilities with full capabilities, including transient and thermal shock, for future DOE programmatic requirements.

OPERATING AND DESIGN PARAMETERS

	GPL-1 Facility		GPL-2 Facility		
	GPL-1	**GPL-1A**	**Loop #2**	**Loop #2A**	**Loop #2B**
Piping	2" Sch. 40 (304 SS)	2" Sch. 40 (304 SS)	6" Sch. 40 (304 SS)	3" Sch. 40 (304 SS)	3" Sch. 40 (304 SS)
Pump	EM Flat Linear Induction Pump	EM Flat Linear Induction Pump	EM Flat Linear Induction Pump	EM Annular Linear Induction Pump	EM Annular Linear Induction Pump
Heater	3.4×10^6 BTU/Hr Output	3.4×10^6 BTU/Hr Output	4.3×10^6 BTU/Hr Output	3.4×10^6 BTU/Hr Output	3.4×10^6 BTU/Hr Output
Flowrate	Up to 200 GPM	Up to 200 GPM	Up to 2000 GPM	Up to 300 GPM	Up to 300 GPM
Temperature	1200°F MAX	1200°F MAX	1200°F MAX	1100°F MAX	1100°F MAX
Pressure	330 psig MAX	330 psig MAX	270 psig MAX	270 psig MAX	270 psig MAX
Cold Trap Flow	2 GPM	2 GPM	5 GPM	5 GPM	5 GPM
Oxygen Purity	1.0 to 5.0 PPM	1.0 to 5.0 PPM	1.0 to 5.0 PPM	1.0 to 5.0 PPM	1.0 to 5.0 PPM
Carbon Purity	0.2 PPM MAX	0.2 PPM MAX	0.2 PPM MAX	0.2 PPM MAX	0.2 PPM MAX
Hydrogen Detection	ON-LINE HYDROGEN METER				
Sodium Chemistry	ON-LINE VANADIUM WIRE STATION				

THERMAL STRIPING TEST FACILITY (TSTF)

The Thermal Striping Test Facility (TSTF) was designed as an addition to the General Purpose Loop-1 (GPL-1) (discussed separately) to perform structural thermal striping tests. The initial test facility design accommodates 4-inch diameter cylindrical specimens up to 14 inches long and wall thicknesses prototypic of LMFBR components. The basic facility can accommodate many other test specimens or component designs.

The TSTF consists of two separate sodium loops supplying multiple test sections with sodium at different temperatures. Within the test section the two sodium streams impinge on a cylindrical rotating specimen outside surface from opposite directions. A thermal baffle within the test section minimizes the mixing between the two sodium streams; this reduces the heat exchanger requirements on recirculation through the two loops. A movable glove box can be positioned over each test section to facilitate test specimen inspection for periodic crack initiation detection.

OPERATING AND DESIGN PARAMETERS

System Fluid	Sodium
Maximum Design Temperature	1200°F
Maximum Design Flow	200 GPM
Maximum Design Pressure	300 psig
Maximum Heat Exchanger Capacity	1 MWt
Sodium Inventory	1000 lb
Maximum Temperature Differential	800°F (1100-300°F)
Specimen Rotation Speed	1-10 Hz

THERMAL-HYDRAULIC TEST FACILITY (THTF)

The Thermal-Hydraulic Test Facility (THTF) is used to obtain steady state and transient data on thermal-hydraulic performance in sodium of electrically heated models of breeder reactor blanket assemblies. The flexible design, performance ratings and operating characteristics of this facility will support a wide variety of component performance, thermal-hydraulic, and Safety Code Validation test programs requiring sodium cooling, electrical heat inputs up to 1500 kW and high speed/high volume automatic data acquisition.

The THTF is one of several parallel test facilities serviced by the General Purpose Loop-1 (GPL-1). The facility consists of an electrically heated test section, inlet and outlet sodium piping, a 1500 kW electrical power supply, a computer controlled data acquisition system satellite and appropriate process and test instrumentation and controls. The facility uses the GPL-1 pumps, etc. to circulate sodium through the test section and uses the GPL-1 Heater/Cooler as a heat dump or heat source. A 61-rod electrically-heated, full-sized model of a breeder blanket fuel assembly is presently being tested in the facility.

During test periods, the facility is operated on a 24-hour per day, 7-day per week basis. This mode of operation maximizes the amount of test data obtained in a given period and thus minimizes project costs by avoiding frequent, costly, and time consuming startup and shutdown operations. Whenever possible, the facility is operated in parallel (piggyback) with other GPL-1 tests to reduce operating costs.

SHUTDOWN CONTROL ROD AND MAINTENANCE FACILITY (SCRAM)

The Shutdown Control Rod and Maintenance Facility (SCRAM) was designed in 1976 for simultaneous testing of three full-size prototype control rod subsystems. The test facility consists of a 110 foot high enclosure, which contains three test stations and a maintenance bay. A twelve inch diameter, segmented, 42 foot long test vessel is installed in each test station. The test vessel was designed to prototypically interface a primary control assembly, driveline and control rod mechanism and can be modified to adapt to future LMFBR primary systems.

The vessels were designed to accommodate predicted LMFBR control rod system flow, temperature, pressure drop and alignment configurations. The vessels are connected to independent flow circuits to provide for independent test operations consistent with program needs. Other salient test facility features include: "built-in" simulated interfaces, including shroud tubes and head/nozzle interfaces; adjustable lateral misalignment and bowing capabilities at several axial locations; ability to measure friction and drag load characteristics; remotely controlled verticality control; and a maintenance bay and storage vessel for conduct of maintenance activities.

Secondary features include: fixed sodium level control through use of overflow vessels; diverse data acquisition systems comprised of analog and digital systems; and a closed loop cold air circulation system for stator cooling.

This facility is used to provide substantial data related to shutdown systems performance and reliability including scram performance and also to determine flow and vibration characteristics.

OPERATING AND DESIGN PARAMETERS

Flow Rate (per vessel)	
Sodium (A&B Vessel)	49.500 lb/hr
(C Vessel)	62.250 lb/hr
Argon Cover Gas	0.0895 scfm
Stator Cooling Air	21.6 lb/min.
Temperature	
Sodium	400-1100°F
Argon Cover Gas	400-1100°F
Stator Cooling Air (Inlet)	60°F
Stator Cooling Air (Outlet)	160°F
Pressure	
Vessel	300 psig
Expansion Tank	150 psig
Sodium Piping	300 psig
Sodium Valves	315 psig
Argon System	80 psig
Stator Cooling Air System	100 psig
Sodium Inventory (per vessel @400°F)	
Vessel	1548 lb
Expansion Tank	166 lb
Sodium Piping and Valves	613 lb
TOTAL	2327 lb

DYNAMIC AND SEISMIC TEST FACILITY (DAST)

Westinghouse has constructed a Dynamic and Seismic Test Facility (DAST) in support of the Clinch River Breeder Reactor Project. This facility was designed to provide a test facility for the current, on-going seismic evaluation of the Clinch River Primary Shutdown System; however, the facility is also capable of seismic testing of other components and subsystems, including sodium-filled piping assemblies. The DAST facility consists of an 80 foot high silo used as a test enclosure. Adjacent to the silo is a 45 foot high reaction mass weighing 265 tons.

For the present Clinch River Program, the facility includes internal test structures supporting a total of eight shakers for inputting vibratory motion. To simulate seismic input at the Clinch River reactor head, three of these shakers are incorporated into a three-dimensional shaker table at the forty-five foot elevation. The result is a facility capable of testing full-scale shutdown systems under simulated seismic conditions inputted at relocatable positions.

Testing in the on-going program has been divided into three phases, utilizing air, water, or sodium as the test medium. Seismic testing of reactor components in a sodium environment has been carried-out through utilization of the General Purpose Loop No. 1 (GPL-1) which is located adjacent to the DAST facility. The ease of startup and shutdown permits one or two-shift operation on most programs.

The test silo-reaction mass facility, combined with the multiple-shaker input and related controls, instrumentation, and Distributed Data Acquisition system, provides a unique testing capability unavailable anywhere else in the United States. The basic facility provides for full-scale testing of complete Shutdown Systems from PWR, LMFBR, or GCFR plants. In addition, the 80 foot high silo located adjacent to an operational sodium test facility permits seismic evaluation of long, full-scale reactor components and piping runs under prototypic sodium conditions.

OPERATING AND DESIGN PARAMETERS

Triaxial Shaker Table 45 Feet Above Ground
- Horizontal Hydraulic Actuator: 10,000 lb. Force, 4 inch Stroke
- Horizontal Hydraulic Actuator: 10,000 lb. Force, 10 inch Stroke
- Vertical Hydraulic Actuator: 14,000 lb. Force, 10 inch Stroke

Five Relocatable Shakers
- Hortizontal Hydraulic Actuators: 5,000 lb. Force, 5 inch Stroke

Hydraulic Power Supplies
- 36 GPM, 3000 psig, 74 HP
- 25 GPM, 3000 psig, 50 HP
- 20 GPM, 3000 psig, 20 HP

50 Ft. High Reaction Mass & Foundation - 534,000 lb.

80 Ft. High 12 Ft. Diameter Silo Test Enclosure

2 Ton Polar Crane

Control Room
- Eight Sets of Shaker Controls
- Input Wave & Sweep Generators
- Instrumentation Consoles
- F.M. Tape Recorders
- Real Time Analyzers & Dedicated Computer

Site Facilities
- Digital Data Acquisition Systems
- Analog & Digital Computer Systems
- Circulating Gas - 36 ICFM, 20 psig Head, 100 psig Static
- Circulating Water - 500 GPM, 300 psig, 180°F; 50 GPM, 2500 psig, 650°F
- Circulating Sodium - 200 GPM, 200 psig, 1200°F
- 1200 KVA, 3 Phase, 480 Volt Substation

ELECTRICAL RELIABILITY TEST LABORATORY (ELREL)

This laboratory is used for reliability testing of plant protection system electronic modules, subsystems, prototypic systems and scram breakers. Additionally, the facility is used for the development and construction of specialized testing and data collection equipment related to plant protection system electronics.

The laboratory is approximately 1200 square feet in area and is serviced by an independent heating and air conditioning system which maintains the required environment for operation of the electronic equipment. The tester configuration currently in use automatically performs 2800 pulse tests on approximately 1000 CRBRP plant protection system modules at 8 hour intervals 7 days per week and records the pass/fail data, the time, date, temperature, relative humidity and line voltage on teletype and magnetic tape. The pass/fail data is also recorded independently on paper tape printers. After manual initiation, the tester automatically performs electrical transfer function measurements on the modules and records the data on teletype and magnetic tape. With the present tester and test system configurations, approximately 11,500 individual measurements are made and automatically recorded every week. The laboratory is equipped with specialized instrumentation such as a digital oscilloscope and a logic analyzer in addition to standard electronic instruments to facilitate maintenance and calibration of the testers and the plant protection system modules.

The tester system is composed of individual tester units which are composed of standardized printed circuit boards wired together to obtain specific input switching, output switching, pulse generation and detection, timing, voltage measurement and alarm functions. The number of devices or assemblies of devices which are to be tested in an identical fashion can range from 1 to 99 for each individual tester unit. The number of individual tester units which may be connected to the main sequencer and data acquisition system is limited only by the requirement that all required pulse tests by all tester units be completed in sequence in less than 8 hours.

All input and output voltages must be within a 12 volt band such as 0 to +12, 0 to −12, +6 to −6, etc. Voltages outside this range must be run through suitable divider circuits.

Voltage or current pulses can be applied in any number or sequence (such as 0 of 3, 1 of 3, 2 of 3, 3 of 3) with setable heights and adjustable pulse lengths. Any number of output pulses may be examined and tested for amplitude and/or maximum propagation delay.

Up to 17 input and output voltages can be measured simultaneously for each device or assembly of devices under test. These measurements can be made at fixed times, with fixed input voltages or from a device response such as a trip. In addition to pulses, waveform voltage or current ramps, and stair-step voltages can be applied to inputs.

OPERATING AND DESIGN PARAMETERS

Input Pulses	1. Up to ±12 volts input 2. Pulse length 0.1 to 99.9 milliseconds or 1.0 to 999.0 milliseconds 3. Propagation delay window, 0.1 to 99.9 milliseconds 4. Output amplitude window, 0 to ±12 volts 5. Current pulses up to ±3 MA available
Input Voltages	1. Ramps from 0 to 5 volts or 5 volts to 0: Rates from 0.1 to 1.0 volts per second 2. Stair-Steps of 1.5, 2.5, 3.5 and 4.5 volts, time at each step adjustable from 0 to 10 seconds 3. Input voltage measurement accuracy ±0.003 volts
Output Voltages	1. Total of input and output signals measured, 17 maximum 2. Measurement accuracy ±0.003 volts
Input Currents	1. Current ramps of 0 to −3 MA ±3 MA to 0, total ramp time of 100 seconds
Data Acquisition	1. Two Estorline Angus PD2064 Data Systems 2. Kennedy 1610/5 Tape Recorder (7 Track) 3. ASR-33 Teletype
120 Volt Power	1. Solid State Regulator 130 amps 120 VAC 60 Hz
Special Equipment	1. Nicolet Digital Oscilloscope 2. Tektronix Logic State Analyzer

MULTI-LOADING TEST FACILITY (MLTF)

The Multi-Loading Test Facility was designed to perform structural tests on high temperature reactor piping components for validation of analytical techniques. The first priority is to perform bend tests on piping components with a combination of external mechanical loading and internal pressure at 1100°F. Test components up to 28 inches in diameter can be accommodated and tests are planned for elbows, tees, and reducers.

The major features of this facility are summarized below:

- Reaction frame to support the test sections with a load actuator system to provide mechanical loading on a variety of test sections.
- Sodium supply to provide an excellent heat transfer medium in the test section, thereby minimizing temperature fluctuations which have a pronounced effect on creep rates in stainless steels at 1100°F.
- Argon cover gas supply and inerting system whereby the test cubicle would be flooded with argon in the event of a sodium leak.
- Instrumentation and control for test section temperature, internal pressure, external force, displacement, strain measurement, sodium and argon supply and automatic data acquisition.

There are no other facilities in the U.S. that can provide the testing capabilities identified above. The automatic control and data acquisition systems allow unattended operation.

OPERATING AND DESIGN PARAMETERS

Test Article Pressure Limit	600 psig
Test Article Temperature Limit	1200°F
Mechanical Force Limit	50,000 lb per load point
Maximum Number of Loading Points	10
Maximum Motion of Test Article	16 inches
Gas Storage Capacity (cover gas and inerting)	29,500 SCF Argon
Liquid Sodium Capacity	4131 lb (900 gal)
System Sodium Flow Rate	Static
Data Acquisition System	200 channel medium speed

HYDRAULIC TEST FACILITY

The Hydraulic Test Facility consists of two, large water loops designed to act either independently of each other, or in conjunction if specific programs require such operation. These two loops are the Thermal Mixing Hydraulic Loop (TMHL) and the Multiple-Purpose Hydraulic Loop (MPHL). The former is a six-inch piping system with flow capability to 2000 GPM at temperatures to 180°F; the latter is a twelve-inch piping system with flow capability to 6000 GPM at temperatures to 180°F. Complete instrumentation, controls, and Data Acquisition Systems are available.

The two loops above also employ the "parallel test section" concept, and have been used on a wide variety of CRBRP, LMFBR, and PWR tasks over the past twelve years. Most recently, their application has included tasks in support of Radial Blanket Flow Orificing, Steam Generator Flow Modeling, CRBRP-PCA Flow and Vibration Studies, and LMFBR Primary Pump Model Testing. Most importantly, however, has been the program effort on the study of Thermal Striping phenomena in the upper internals region of the CRBRP primary vessel. These two facilities have proven to be exceptionally useful in evaluating thermal striping behavior; and they together represent an operational facility ready to extend the experimental scope and knowledge of this critical phenomena.

Through either concurrent or independent operation, the two facilities provide a "cost-sharing" option to minimize the operating costs of multiple programs being carried-out concurrently in the Hydraulic Facility. The ease of startup and shutdown permits one or two-shift operation on most programs.

OPERATING AND DESIGN PARAMETERS

	TMHL	MPHL
Maximum Design Temperature	180°F	180°F
Maximum Design Pressure	200 psi	200 psi
Design Basis	Power Piping Code ANS B 31.1 0	Power Piping Code ANS B 31.1 0
System Fluid	Water	Water
Maximum Flow Rate	2000 gpm	6000 gpm
Pump Head	100 psig	100 psig 5500 gpm
System Capacity	3400 gal	1500 gal
Filtration	Full Flow Filter	Full Flow Filter
Piping (Type 304 SS)		
• Main System	6 inch	12 inch
• Auxiliary	3 inch	1, 4, and 6 inch
Flow Measurement		
• Main Stream	6 inch Orifice	12 inch Orifice
• Auxiliary	3 inch Orifice	1 and 4 inch Flowmeters 6 inch Orifice

CREEP RATCHETING TEST FACILITY (CRTF)

The Creep Ratcheting Test Facility was designed in 1973 for thermal transient testing of full size reactor piping components. The initial design is specifically aimed at a study of creep ratcheting in the FFTF/IHX inlet and outlet nozzles, however, the basic facility has a wide range of flexibility on test section design. The first tests were performed with a worst-case FFTF plant transient at hold temperatures and internal pressures similar to those in FFTF. Subsequent tests are being performed at more severe conditions to provide data for validation of analytical techniques for creep ratcheting strain accumulation.

Argon is used to simulate sodium transients and was chosen over nitrogen to eliminate spurious test results associated with nitriding. No other argon facility is available to provide the required test conditions and high speed data acquisition. All of the test nozzles are welded into one full scale test section simulating the FFTF/IHX. Even if another test facility were available, the cost and schedule impact of preparing, shipping and re-installing this massive test section would be prohibitive.

The facility is designed to simulate plant transients by passing a high flow rate of inert gas over the critical surfaces of the test article. Flow area is controlled by internal baffles and flow rate is controllable from 2000 to 100,000 lb/hr. The gas supply system includes a cryogenic pump, vaporizer and storage tubes with the main piping loop fabricated with four to six inch pipe. Test temperatures up to 1200°F and test pressures up to 600 psig can be accommodated. The test section is located in a blast resistant test cell with a remote control room and a high-speed data acquisition system that can handle up to 200 channels.

The automatic control and data acquisition system allow unattended operation of the facility except during the periodic transients, thereby reducing operating costs.

OPERATING AND DESIGN PARAMETERS

System Fluid	Gaseous Argon
Maximum Design Temperature	1200°F
Design Pressure	600 psig
Design Basis	B 31.1.0 Power Piping Code
System Flow Rate	100,000 lb/hr
Pipe Material	ASTM-A-312, Type 304 and 316 Stainless Steel
Pipe Sizes: Main System	6-inch Sch 40
Gas Storage	300,000 SCF at 2500 psig
Cryogenic Liquid Storage	2600 gal
Vaporization and Pressurization	800 SCMF to 2500 psig
Total System Inventory	400,000 SCF
Data Acquisition System	High Speed Digital Data Logger

DISTRIBUTED DATA ACQUISITION SYSTEM (DDAS)

The Distributed Data Acquisition System (DDAS) is used to acquire, log and process test data from major test facilities at the Waltz Mill site. The system provides real-time, interactive data acquisition and display capabilities for monitoring steady state and/or transient test operations and off-line computational capabilities for reduction, analysis and presentation of test data and results

The DDAS consists of four remotely located mini-computer controlled analog/digital measurement subsystems connected to a central mini-computer facility. The remote satellites provide the data signal measurement equipment, sufficient input/output and storage equipment for limited software development, and data processing for effective operator or computer interactive real-time monitoring of test operations. The central facility provides the program development resources, mass storage equipment and high speed input/output equipment for development of applications and utility software programs and for off-line reduction and analysis of test data.

The facility is available on an open basis to users, and supports multi-programming, interactive, real-time software development and data processing. A wide variety of data conversion, characterization, plotting and analysis programs are utilized. These programs form the basis for development of the applications programs packages to meet specific test requirements. Satellite operation during test periods is handled by regular facility operations personnel.

The facility (Central & Satellites) is presently serving the Creep Ratcheting Test Facility (CRTF), Multi-Loading Test Facility (MLTF), Thermal-Hydraulic Test Facility (THTF) and the Shutdown Control Rod and Maintenance Facility (SCRAM). The Few Tube Model Steam Generator Test will also be serviced by the THTF Satellite. In addition to the data acquisition and processing work performed for these programs, several other projects utilize the Input/Output, computational, graphics and data storage and retrieval capabilities of the Central Facility.

OPERATING AND DESIGN PARAMETERS

- Acquire analog data from multiple remote sensors at sampling rates consistent with nature of the test.
- Log raw data for permanent storage on magnetic tape.
- Provide real-time displays in engineering units of key test conditions, test data and calculated parameters for use by operators in monitoring set-up and performance of tests.
- Assist operators through pre-programmed instructions in proper set-up and performance of tests.
- Alert operators of out-of-limit conditions by monitoring selected parameters.
- Provide calculational aids and tabular and graphical hard copy output capabilities for real-time assessment of test results and/or off-line processing and analysis of test data.

Materials Test Loops (MTL-2, -3)

The materials test loops are designed to provide information on the metallurgical behavior of LMFBR system materials under prototypic corrosion and deposition systems. They are therefore designed to produce the same corrosion and deposition effects as full-scale primary and secondary LMFBR sodium circuits. Position-dependent effects are obtained at the end of each run by evaluating sections of loop piping. Time-dependent effects are studied using test coupons periodically removed from hot leg and cold leg test sections - this can be done without interrupting sodium flow by means of an inert atmosphere drybox system.

Historically, the facilities were used to complete a statistically designed test matrix of 42 campaigns which resulted in basic design data on corrosion wastage and deposition effects on the thermal-hydraulic behavior of the FFTF and CRBRP systems. Current work involves exposure of special component materials to clarify their sodium compatibility behaviors. More specifically, MTL-2 and MTL-3 are used to obtain data on the chemical, physical, and mechanical properties of coatings and other wear-resistant surfaces considered for FFTF applications, specifically those data required on sodium compatability and mechanical integrity.

This facility originally included four loops in line with a traveling, dry argon-filled drybox over them. The drybox was moved from one sample tank flange to another for withdrawal of exposed hot leg and cold leg coupons. Two of the loops have been modified to perform priority work of other types and the drybox has been removed for service on another facility.

The loops are constructed of 0.5" pipe and can be torn down for evaluation and rebuilt on an approximate 30-day cycle.

MTL-2/3 were designed for 2.7 gpm maximum flow, a maximum sample section temperature of 1325°F, and a ΔT of 300°F. Oxygen level of the sodium is maintained between 0.5 and 1.0 ppm by continuous cold-trapping and is measured by a vanadium wire equilibration device.

Sodium compatibility testing of specialty (hard facing, etc.) component materials for FFTF and CRBRP design groups is done on a "need" basis. Current base loading is sodium compatibility testing of fuel subassembly duct load pad materials for use with advanced fuel cladding and duct alloys.

OPERATING AND DESIGN PARAMETERS

T_{Max}, Hot Leg	1000 to 1325°F
T_{Min}, Cold Leg	700 to 1200°F
Nominal Na Flow	2.3 gpm

Configuration Features

- Hot and cold leg sample exposure sections (Isothermal)
- Cold trap with economizer
- Vanadium wire module for oxygen measurement
- Main flow economizer (LMFBR/IHX simulator)
- Hot leg heater section (12 KW, radiant source)
- Cold leg air dump heat exchanger

Interstitial Element Transfer Facility (ITF)

The ITF loop provides a test system used to investigate the transfer phenomena of interstitial alloying elements (carbon, nitrogen, silicon, boron) in a dynamic polythermal system. It approximates an operating reactor in terms of thermal and mass fluences, surface area ratios, and materials of construction.

The oxygen concentration in the sodium is controlled using a cold-trap and monitored directly by a vanadium wire equilibration device, an electrolytic oxygen meter, and indirectly by a hydrogen meter. The carbon concentration in the sodium is characterized using a standard tab exposure chamber.

This facility (unique in the U.S.) was designed as a corrosion potential simulation of the FFTF primary system. The three specimen test locations simulate the FFTF hot channel, the FFTF average, and the FFTF low fuel subassembly sodium temperature heat rise rates. The hot channel location temperature rise rate is 350°F over 30 inches. Corrosion kinetics can be quantitatively evaluated at typical LMFBR reactor core locations in these test sections. Deposition effects can also be studied in other ITF locations which simulate other positions in an LMFBR primary sodium system circuit.

ITF was used to set the fuel pin cladding corrosion wastage allowances for both FFTF and CRBRP design. For the same applications, the rate of carbon and nitrogen loss from 20% cold worked Type 316 stainless steel were established, providing baseline date for the evaluation of mechanical properties effects. Currently, ITF is being used to evaluate the corrosion behaviors of advanced alloys for LMFBR fuel cladding, fuel subassembly ducts, and core structural applications.

OPERATING AND DESIGN PARAMETERS

T_{Max} (Normal)	1250°F
T_{Max} (Max Design)	1400°F
Flow Rate (Normal)	6 gpm
Flow Rate (Max Design)	8 gpm

Carbon/Nitrogen Equilibrium Loop (CEL-2)

CEL-2 is used to provide experimental information and analytical relationships for FFTF and CRBRP design use so that the rate and direction of carbon and nitrogen transfer may be predicted. Associated mechanical property changes of structural materials, within liquid metal circuits representative of primary and secondary reactor plant systems, can then be estimated.

The principal function of a CEL is to supply sodium of known temperature, carbon activity, and oxygen content to a large flow-through sample tank. Thin foil samples of the structural materials under study are allowed to equilibrate with the sodium in the sample tank, and are then analyzed for carbon. Control of oxygen level is accomplished by means of a cold trap, and determination of oxygen level is by the vanadium wire equilibration method. Activity of carbon in the sodium is adjusted by means of an encapsulated carbon source, the temperature of which is manipulated to control the rate of carbon introduction. Carbon activity is monitored by the standard tab equilibration technique.

This is the only facility of its type in the U.S. It offers a means of studying carbon migration kinetics over a wider range of carbon concentrations in sodium than is practical in normal sodium circuits. Thus, evaluation of off-normal events is possible with minimal risk. Furthermore, because of the nature of the specimens used, large numbers of determinations can be made quite rapidly in comparison with the time demanded by normal sodium loop techniques.

OPERATING AND DESIGN PARAMETERS

T_{Max}	1400°F
Flow Rate (Normal)	2 gpm
Flow Rate (Max)	3 gpm

Self-Welding System (SWS-1)

The SWS loop is used to provide experimental information on the self-welding characteristics of reactor materials in sodium with relevance to the component configurations and conditions anticipated in the reactor system.

System operational parameters include primary sodium flow of two gpm at a maximum temperature of 1200°F and a ΔT of 300°F. Oxygen level of the loop is kept between 0.5 and 1.0 ppm using a cold trap and is measured by a vanadium wire equilibration device. Three sets of test chambers are connected in parallel. A total of 9 test chambers are normally used. Sodium flow through the test chambers is maintained between 0.25 - 0.5 gpm.

Specimens consist of two members with contact areas designed to simulate the component under investigation. After a period of sodium wetting, the two members are brought into contact and loaded hydraulically to the desired Hertzian level and held under load at the prototypic temperature, sodium purity, etc. At the conclusion of the contact period, separation forces are measured in tension or in torsion as appropriate.

SWS-1 is used to collect data in support of FFTF and CRBRP design (valves, CRDM, IVHM, etc.) and under the LMFBR Base Technology programs.

OPERATING AND DESIGN PARAMETERS

T_{Max}	1200°F
Na Flow	2.0 to 5.0 gpm
Loop T	350°F (Max)

Sub-Component Test Facility (SCTF)

The SCTF is a general purpose sodium test facility which was designed to bridge the gap between the small mass transfer systems (3-6 gpm) and the large component test loops (200-2000 gpm). In its present form, the facility consists of two separately pumped, high-flow rate (up to 100 gpm), isothermal loops which operate at 750°F. The loops share a common dump tank and a Vandium Wire Equilibration Device which monitors oxygen concentration in the sodium. Separate cold traps control the impurity content of the sodium in each loop. Liquid metal pressure transducers are used to measure the pressure drop across each test section.

To the basic facility configuration, test assemblies are attached and sodium performance data is acquired. The data is then directly applicable for engineering design use.

The current test assembly was designed to perform the CRBRP Lower Inlet Module (LIM) Orifice Life Test. The objective of this test is to determine the corrosion/erosion behavior of prototypic orifice plate configurations under prototypic hydraulic conditions. The test sections consist of a stack of four prototypic orifice plates and spacers rigidly clamped and enclosed in a flow through sleeve.

OPERATING AND DESIGN PARAMETERS

	Loop A	Loop B
Loop Temperature	750°F	750°F
Na Flow	100 gpm	60 gpm
Velocity at test section	30 fps	30 fps

Sodium Friction and Wear Testing Rigs (SWEATER-1, -2, -3)

These loops measure the friction coefficients and wear rates of various reactor component materials combinations under anticipated operating conditions, and provide reactor designers with quantitative data and experience factors necessary to successfully design reactor component interfaces. Testing has been in sodium, argon cover gas, air, and helium in support of the LMFBR program and GCFR development.

Three Sodium Wear Test Rig (SWEATER) facilities have been constructed. Each facility consists of four test vessels connected in parallel to a sodium supply loop. The sodium supply loop is essentially an isothermal, EM pumped system containing approximately 50 pounds of sodium, capable of supplying a total of 5 gpm at temperatures up to 1250°F to any two parallel test vessels. Sodium impurity levels are controlled by continuously circulating 0.2 to 0.5 gpm through a cold trap at approximately 260°F. An on-line electrochemical oxygen meter and vanadium wire equilibration device are provided to characterize the sodium. Each vessel is provided with an auxiliary line to permit independent filling or draining while the remaining loop and test chambers are in operation.

Each test vessel uses a dual interface, diametrically opposed loading technique with vertically reciprocating motion. FM magnetic tape data acquisition and computer data reduction are used in processing the friction data. These facilities are unique in the U.S.

OPERATING AND DESIGN PARAMETERS	
T_{Max}	1250°F
Flow Rate (Normal)	2 gpm
Flow Rate (Max)	3 gpm

Thermal Cycling Facility (TCF-1, -2, -3)

These facilities are used to test materials damage mechanisms and kinetics due to thermally-induced creep and fatigue strains in outer surface regions. The tests are conducted under prototypic thermal/mechanical and sodium purity conditions.

TCF-1 consists of a sodium supply loop, two sample quench tanks, and an industrial robot system used to transfer the 2" Th. × 4" × 4" specimen from one tank to the other. The robot system can be pre-programmed to obtain the desired time-temperature cycle.

TCF-2 is designed to simultaneously cycle three test specimens in a triple fixture attached to the top of a large volume sodium pot, which contains approximately 1100 lb of static sodium at 300°F. The three test fixtures, fabricated from stainless steel, operate independently of each other. Each test fixture utilizes its own pneumatic drive system and is capable of operating in both a manual or automatic mode. Also, each contains a furnace section where the test specimens are heated to temperatures up to 1200°F in an atmosphere of argon and sodium vapor, then transferred vertically to the sodium pool for quenching. The facility is capable of performing thermal down shocks (argon to sodium) over the temperature range 1200→301°F.

TCF-3 is an upgraded version of TCF-1, permitting improved cycling versatility and the use of larger specimens. Owing to mechanical improvements, both more rapid cycling and greatly enhanced equipment availability are realized.

All three facilities are served by sodium loops with sodium purity and oxygen monitoring capabilities. The materials of construction are Types 304 and 316 stainless steels.

All three facilities are involved in the LMFBR Base Technology and CRBRP programs on effects of thermal cycling (high strain, low frequency) and thermal striping (low strain, high frequency) on sodium system component performance. In addition, there has been limited work of the same type sponsored by the Solar Energy Research Institute.

OPERATING AND DESIGN PARAMETERS

TCF-1	
Drive System	Pneumatic Transfer Between Tanks
Sample Tank Sizes (2)	6" Dia. × 24" H.
T_{Max} (Tank #1)	1100°F
T_{Min} (Tank #2)	400°F
Na Flow	2 gpm
Na Inventory	~500 lb
Max Cycle Speed	1.5 Min
TCF-2	
Drive Systems (3)	Vertical Pneumatic
Sample Tank (1)	One ~24" Dia.
Furnaces (3)	3" I.D. × 12" L; Radiant Elem
T_{Max} (Furnace)	1200°F
T_{Min} (Na Tank)	300°F
Na Flow	2 gpm
Na Inventory	1100 lb
Max Cycle Speed	~1 Min
TCF-3	
Drive System	Vertical/Rotation Pneumatic
Sample Tank Sizes (2)	8" Dia. × 28" long
T_{Max}	1200°F
T_{Min}	400°F
Na Flow	5 gpm
Na Inventory	~200 lb
Max Cycle Speed	~1.0 Min

In-Sodium Mechanical Property Test Systems (MPS-1, -2, -3, -4, RCLC-M)

The primary objective of these facilities is development of baseline data on the effect of sodium corrosion on the mechanical behavior of austenitic stainless steels and nickel-based alloys. Creep, creep-fatigue, fatigue, simple tension, simple compression, and stress-relaxation data are obtained under corrosion conditions corresponding with those in LMFBR sodium circuits.

Because of the design of the facilities, tests in air, inert gases, liquid metals other than sodium, and a variety of other liquid media are possible. However, the urgent requirements of the LMFBR development program have to date excluded work in media other than sodium.

MPS-1, -2, -3, -4, and RCLC-M are individual facilities consisting of three major components: (1) mechanical property testing machine(s), (2) test chambers and (3) sodium supply loop. MPS-1, -2, and RCLC-M have Creep Machines for in-sodium creep rupture testing. MPS-3 has two Mechanical Test Systems (MTS) closed-loop servo-hydraulic fatigue machines, each with its own test chamber. MPS-4 consists of a single MTS fatigue unit and test chamber for in-sodium fatique testing. The sodium loops are designed to supply LMFBR prototypic sodium to the test chambers. Sodium temperature varies between 1000 and 1300°F depending on the desired test conditions. The oxygen level in the loop sodium is maintained at 1.5-2.0 ppm by continuously cold trapping 5% of the main loop flow. The oxygen and carbon concentrations are periodically measured by vanadium wire equilibration devices. Maximum mass flow rate in each loop is 2.5 gpm.

Present work loads relate to the LMFBR Base Technology program on materials and structures as well as to specific FFTF and CRBRP data needs. Materials included in the several test matrices are Types 304 and 316 stainless steels and Inconel 718. These facilities are unique in the U.S.

OPERATING AND DESIGN PARAMETERS

The sodium supply loops for all four facilities are basically slightly scaled up versions of materials test series loops (MTL's). In this series, the isothermal hot leg section is replaced by a mechanical test chamber into which the mechanical test machine grips are introduced via stainless steel bellows. Sodium is directed past the test specimen gage length downward through an orifice designed to give the desired flow velocity.

$T_{(Max)}$	1325°F
Loop $\Delta T_{(Max)}$	350°F
Sodium Velocity at Sample Surface (Max)	20 fps

Mechanical loading machines are of standard types. The creep-fatigue machines are computer-programmed MTS units of 22 KIP capacity.

Fission Product/Thermal Gradient Test Loop (FPL/TGL)

This facility was originally constructed to examine the efficiencies with which radioactive species can be removed from LMFBR sodium circuits by hot traps and cold traps. Some work was also done on development of magnetic traps using Westinghouse funds. Early data were developed on the distribution of radioactive deposits to be expected in the FFTF primary system.

More recently, a small water/steam loop (TGL-1) was added to permit detailed evaluation of the temperature distributions in duplex Croloy (2 1/4 Cr-1 Mo) steam generator tubes under steady-state and off-normal conditions.

FPL-2 includes a primary loop, two cold-trapping circuits, and a hot-trapping purification circuit. The primary loop, fabricated of 1/2-inch OD stainless steel tubing, consists of a 2 gpm electromagnetic pump, 20 KW immersion heater, electromagnetic flowmeter, cooler, valves, sump tank, and expansion tank. The cold and hot trap loops are identical except for the internals of the traps. Each purification loop consists of the trap, economizer, electromagnetic flowmeter, and valving. The loop is also equipped with a vanadium wire equilibration device and oxygen and hydrogen meters.

A steam/water supply system has been built adjacent to FPL-2, for steam generator tube tests. It consists of the following components in series: water supply demineralizer bed; water supply tank with preheaters and argon purge; hydraulic pump; main heater; test specimen with bypass line; dump heat exchanger; pressure regulator; flowmeter; drain.

The facility is being used to provide thermal design data for the ANL/Ⓦ Tampa Large Scale Breeder Reactor (LSBR) steam generator design and development contract.

OPERATING AND DESIGN PARAMETERS

Flowing Sodium Inventory	40 lb
System Tubing Size	1/2 inch O.D. 304 SS
Maximum Operating Temperature	1200°F
Maximum T Around Loop	200°F
Immersion Heater Power	20 KW
Number of EM Pumps	2 (1 in Main Loop, 1 in Sub-loop)
Nominal Flow	2 gpm
Purification Systems	2 Cold Traps, 1 Hot Trap
Purity Verification Systems	On-Line Oxygen and Hydrogen Meters Wire/Tab Equilibration Module Bypass Sampling
Steam/Water Supply System:	
Maximum Operating Temperature (steam)	850°F
Maximum Operating Pressure	2200 psig
Maximum Water Flow	45 lb/hr

Mechanical Properties Laboratory

The purpose of this laboratory is to provide a capability for mechanical property testing of materials for ambient and elevated temperature service.

The laboratory is equipped with sixteen high precision, lever arm creep units used in support of high temperature design studies of Types 304 and 316 stainless steel, Alloy 718, SA 516 and modified 9 Cr-1 Mo steels.

Six of the units produce multiaxial stress conditions in the specimen by a combination of a dead load and internal pressure. This system is unique in that axial and circumferential extensions are measured simultaneously as a function of time. Temperature is controlled to $+3°F$ and creep is measured to 5×10^{-5} inch.

Ten of the creep units produce axial stress in the specimens and have similar temperature and strain measuring precision to the multiaxial units.

Three low-load creep rupture units, each with eight test stations, are also used for creep testing. These units have similar temperature and creep measuring precision to the other units.

A 60,000 psi tensile test unit is used for routine tension and compression tests on materials and has been used to determine deflection of full-size duct components under various loading conditions. It has also been used to evaluate crush strength of various types of insulation.

The creep testing area is currently being equipped with a microprocessor system whereby all strain and temperature sensors will be continuously monitored and data stored on disc for subsequent analysis.

The laboratory has three Mechanical Test System (MTS) universal testing machines that can be used for tensile/compression and fatigue testing. Two of the units are of 60,000 lb capacity and one is 200,000 lb capacity. Currently one of the units is being used to evaluate the high temperature fatigue performance of Types 316 stainless steel. The second unit is being used to evaluate the wear characteristics of duplex 2 1/4 Cr-1 Mo steam generator tubing in a cyclic mode at elevated temperatures. The third unit is also used for fatigue testing of modified 9 Cr-1 Mo and for tensile testing of a variety of materials. All units can control temperature to $\pm 3°F$ with resistance, induction or radiant heating furnaces. In addition, the units are equipped with the necessary X-Y recorders, strip chart temperature recorders and oscilliscopes.

This laboratory produces data for essentially all LMFBR Base Technology Tasks in which materials mechanical properties testing is involved. Several CRBRP design support tasks are involved as well.

In addition to standard specimen testing, friction testing of steam generator tubing internal interfaces, mechanical behavior of fuel subassembly ducts and fuel pin grids, and a large variety of other component mechanical tests are performed.

Fabrication Laboratory

This laboratory provides facilities for fabrication and assembly of components associated with fuel assemblies for both in and out-of-reactor testing. The development of welding and assembly techniques for honeycomb grid spacer systems, in various materials, to RDT Specifications, and the design of assembly and welding fixtures, are two significant areas of work scope undertaken in the Fabrication Laboratory.

Major items of equipment in the laboratory include:

- 75 KVA resistance welding machine
- 50 KVA resistance welding machine
- Wire wrapping machine
- Assembly and welding fixtures for honeycomb grid and grid to hexagonal duct welding.

The following is a list of typical tasks completed by the Fabrication Laboratory in recent years.

- Articulated Control Rod Assembly
- Grid-Type Driver Fuel Subassemblies for WSA Series Testing in EBR-II (37-Pin Size)
- As above, 19-Pin Size
- Wire-wrapped Radial Blanket Fuel Subassemblies for WBA Series Testing in EBR-II (7-Pin Size)
- Grid-type Driver Fuel Subassemblies for FFTF (217-Pin Size)
- Various Mechanical and Thermal/Hydraulic Instrumented Test Articles of the above types.

Flow test articles of 37-pin and annular (90 pin of two sizes) gridded configurations are being produced for the (U, Pu) Carbide fuel test program in FFTF. Irradiation test hardware for the same program is also in process.

H. UNIQUE FACILITIES AND OTHER RESOURCES

As a consequence of its commitment to and extensive involvement in virtually all of the fields of advanced energy technology, Westinghouse is in a position to bring to bear a wide range of existing facilities to support the various elements of the First Wall/Blanket/Shield Engineering Test Program. Figure H-1 schematically depicts the spectrum of corporate resources available.

Several of the facilities available to the program offer unique capabilities to meet the specific requirements of the surface heat load and related effects test program element. The total range of facilities available offer future flexibility in terms of testing with alternate coolants and the combination of effects. Table H-1 is a concise listing of this array of facilities with specific and potential relevance to the test program element. All of the facilities listed are located in the Pittsburgh area, which facilitates communication and coordination by a central planning and engineering support organization (Fusion Power Systems Department). The following sections provide a description of each major facility and its proposed or potential role in the program.

H.1 THE ELECTRON BEAM SURFACE HEATING TEST FACILITY (ESURF)

In recognition of the need for near term surface heating experimental capability for the engineering development of a wide range of fusion device components, Westinghouse has assembled a flexible surface heat load facility, based on an electron beam source, at its Research and Development Center. The facility became operational in its present configuration in 1980 and was used to test water cooled cathodes in the context of the negative ion source development program performed for the Brookhaven National Laboratory.

Westinghouse is presently negotiating with the MIT Plasma Fusion Center to perform magnetic divertor collector experimental studies using this facility.

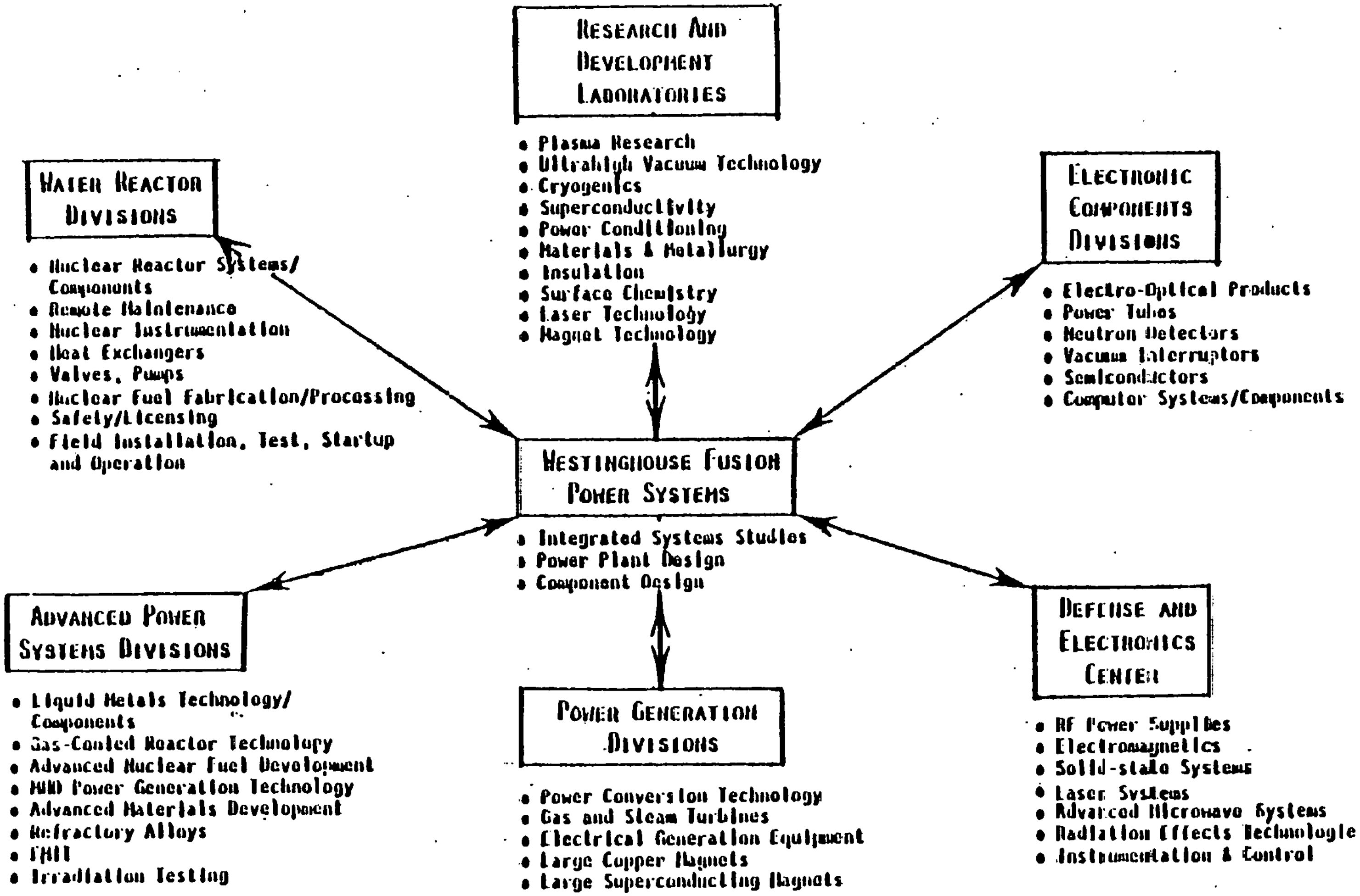


FIGURE II-1. CORPORATE RESOURCES AVAILABLE TO SUPPORT WESTINGHOUSE FUSION PROGRAMS

TABLE H-1
SURVEY OF WESTINGHOUSE FACILITIES SUITABLE FOR FUSION HARDWARE DEVELOPMENT AND TEST

FACULTY DESIGNATION	LOCATION	APPLICATION-DATA GENERATED	COMMENTS/FUSION RELEVANCE
Electron Beam/High Surface Heat Flux Test Facility	W R&D Center Churchill, PA	Thermal/Hydraulic/Mechanical Response of Targets exposed to high heat flux beams in a vacuum environment. DC flux simulation using rastering over areas ranging up to 30 cm by 30 cm in size with a 36 kW beam coupled to a high pressure water loop for target cooling.	Dedicated to fusion research - used for scaling experiments and screening tests. Test section is in a vacuum chamber, beam may be electronically cycled for pulse testing.
Light Ion Beam Test Stand (LIBTS)	W R&D Center Churchill, PA	Characterization of material performance under ion beam and atomic beam bombardment. Test section is in a vacuum chamber and targets may be cooled by a high pressure water loop. Provides a 150 keV beam at 1.5 kW.	Dedicated to fusion research. Off-line analysis of implanted test targets via a secondary ion spectrometer facility.
High Temperature Liquid Lithium Facility	W R&D Center Churchill, PA	Fluid dynamics and heat transfer in flowing lithium systems: lithium aerosols, jets, sprays, free surface flow, wetted surfaces and screens. Can be used with the electron beam test stand, plasma-arc heater or radiant heating sources.	Dedicated to fusion research.
Materials Examination Facility	W R&D Center Churchill, PA	Destructive and nondestructive metallurgical examination using such equipment as an electron beam microanalyzer, scanning electron microscope, scanning transmission electron microscope, Auger electron spectroscopy techniques, light microscopy and photomicrography.	Available as required to provide pre-test, during test, and post-test characterization of test pieces.
General Purpose Boiling Heat Transfer and Two Phase Flow Loop	W R&D Center, Churchill, PA	General Purpose Heat Transfer Studies using R11, R114, hydrocarbons or water. Particularly used for cross-flow boiling studies simulating nuclear steam generator conditions. Maximum conditions: 250°F to 550°F, 180 psia-1100 psia, at 20,000 lbm/hr.	Available to support thermal/hydraulic studies on water-cooled fusion reactor components.
High Power CO_2 (COFFEE) Laser Facility	W R&D Center, Churchill, PA	High power laser development facility - can be used for intense target heating.	Available for possible future fusion R&D.

TABLE H-1 (CONTINUED)
SURVEY OF WESTINGHOUSE FACILITIES SUITABLE FOR FUSION HARDWARE DEVELOPMENT AND TEST

FACULTY DESIGNATION	LOCATION	APPLICATION-DATA GENERATED	COMMENTS/FUSION RELEVANCE
High Energy Ion Bombardment Simulation Facility (HEIBS)	University of Pittsburgh	Surface and bulk damage studies to simulate high energy neutron effects. Simultaneous bombardment of targets with high energy multiple ion beams. (< 2 MeV light ions = He, H, etc.) (≤ 60 MeV heavy ions). Static (post bombardment) and dynamic (in situ) mechanical measurements. Important properties being examined and techniques utilized include modulus, internal friction, uniaxial tension, fatigue, and stress relaxation.	Jointly owned and operated by W and the University of Pittsburgh. Co-Supported by OFE DAFS Task Group activities and the National Science Foundation.
Arc Heater Laboratory (HX-10)	East Pittsburgh, PA	General purpose facility permitting tests of arc heater applications (500 kW to 3.5 MW per unit). Sodium and water heat transfer loops are available. The heater may be operated with air, argon, H_2-He mixtures or chemical feedstock gases. Targets may include liquid metal droplets, solid particles or fluid-cooled surfaces. For a 500 kW unit it is estimated that a surface heat flux of 3-5 kW/cm^2 can be attained over a target area of 45-80 cm^2.	Latest application tested was an arc heater silicon production process for JPL involving heating of liquid sodium sprays by three 500 kW units. Has potential for fusion component testing. Requires fabrication of a test chamber which can be mated with the existing power and hydraulic loops.
"A&B" Low Flow/High Pressure Hydraulic Loops	W PWR Test Engineering Laboratory, Forest Hills, PA	Full Scale valve tests, material corrosion-erosion, corrosion product release and transport 150 gpm at 2400 psi.	Available for fusion blanket coolant system component evaluation/qualification (future applications).
"D" Loop - Medium Flow High Press	W PWR Test Engineering Laboratory, Forest Hills, PA	Operational studies of large PWR core components 4400 gpm at 2400 psi. Services a 25" x 38' long test vessel.	Available for fusion blanket coolant system component and subsystem thermal hydraulic evaluation/qualification. (possible future application)
"G" Loop - Engineering Core Cooling System Facility	W PWR Test Engineering Laboratory, Forest Hills, PA	Test data for analysis of LOCA (Loss-of-Coolant - Accident) high pressure simulation system rated for 2000 psi and 650°F.	Available for total fusion primary coolant loop simulation and study of loss of coolant situations. (possible future application)

TABLE H-1 (CONTINUED)
SURVEY OF WESTINGHOUSE FACILITIES SUITABLE FOR FUSION HARDWARE DEVELOPMENT AND TEST

FACULTY DESIGNATION	LOCATION	APPLICATION-DATA GENERATED	COMMENTS/FUSION RELEVANCE
"E" Loop - Low Flow, Low Pressure Hydraulic Facility	W PWR Test Engineering Laboratory, Forest Hills, PA	Flow and vibration studies with component mockups (including plastic models for visual observation) 1000-2000 gpm.	Available for fusion coolant system flow oscillation and vibration studies using inexpensive mockups.
"Flecht-Set" Emergency Core Cooling Facility	W PWR Test Engineering Laboratory, Forest Hills, PA	Low-pressure (0-60 psia) facility designed to provide experimental data on the influence of system effects on Emergency Core Cooling Systems during the reflood phase of Loss of Coolant Accident.	Available for total fusion primary coolant loop simulation (including steam generator modules, valves, pumps, etc.). (possible future application)
Mechanical Testing Laboratory	W PWR Test Engineering Laboratory, Forest Hills, PA	Full scale mechanical and vibration tests on plant and reactor components to prove reliability.	
"J" Loop - Delayed Departure from Nucleate Boiling Heat Transfer Facility	W PWR Test Engineering Laboratory, Forest Hills, PA	Instrumented pressurized water test facility for verifying DNB phenomena during transient and steady state system operation. Will accept a full size single loop simulation of a four-loop PWR reactor system 2500 psia at 650°F, flow rates to 450 gpm	Available for testing of large scale fusion coolant system components with internal heating (via electrical resistance heating). (possible future application)
"H" Loop - High Flow Hydraulic Test Facility	W PWR Test Engineering Laboratory, Forest Hills, PA	Permits use of 1/2 scale reactor models and full-scale fuel assemblies for conducting mixing studies, flow distribution studies and similar low temperature, low pressure hydraulic tests Maximum flow rate = 14,000 gpm ΔP: 120 psi, water temperature range: 70°F-200°F	Available for fusion component scale experiments for complex flow distribution studies. (possible future application)
Lithium Facility for Fusion-Related Experiments (LIFE)	W ARD - Waltz Mill, PA	Basic corrosion and deposition information for materials in contact with lithium for fusion applications. Lithium system operating experience development, impurity monitoring and control development.	Dedicated to fusion research. This loop is about 70% complete - to be operational in June 1981

TABLE H-1 (CONTINUED)
SURVEY OF WESTINGHOUSE FACILITIES SUITABLE FOR FUSION HARDWARE DEVELOPMENT AND TEST

FACULTY DESIGNATION	LOCATION	APPLICATION-DATA GENERATED	COMMENTS/FUSION RELEVANCE
General Purpose Sodium Facility #1 (GPL-1) (Two Loops)	W ARD; Waltz Mill, PA	Max. flowrate: 200 gpm, temperature: 1200°F, pressure 330 psig.	Secondary side water loop in this facility proposed for large scale FW/B/S testing (has a 2 MWt heat rejection capacity).
o Steam Generator Test Facility (SGTF)		LMFBR steam generator performance data under both steady state and transient conditions	
o Thermal Striping Test Facility (TSTF)		High Temperature Structural Design	
o Thermal-Hydraulic Test Facility (THTF)		Thermal/Hydraulic and heat transfer data for design and safety code validation (electrically heated core components)	
General Purpose Sodium Facility #2 (GPL-2) (Three Loops)	W ARD; Waltz Mill, PA	Max. Flowrate: 2000 gpm; temperature: 1200°F at 270 psig.	Possible future application to lithium systems through similitude.
o Shutdown Control Rod and Maintenance Facility (SCRAM)		LMFBR Safety/Reliability Program Data. Subsystem and Component Performance Data.	
Dynamic and Seismic Test Facility (DAST)	W ARD; Waltz Mill, PA	LMFBR Safety/Reliability Program Data for Shutdown Systems (Soidum-filled piping and assemblies).	Possible future application for the seismic qualification of fusion systems components
Electrical Reliability Testing Laboratory (ELREL)	W ARD; Waltz Mill, PA	LMFBR Plant Protection System Safety and Reliability Data-Electronic module lifetime testing.	Possible future application to the development of a reliability data base for fusion system power supplies and electronics.
Multi-Loading Test Facility (MLTF)	W ARD; Waltz Mill, PA	High Temperature Structural Design-Structural tests on sodium-filled piping and components at high temperature	Possible future application to fusion reactor components.
Creep Ratcheting Test Facility (CRTF)	W ARD; Waltz Mill, PA	High temperature structural design-creep ratcheting strain accumulation - test temperatures: 1200°F.	Possible future application to fusion reactor components.

TABLE H-1 (CONTINUED)
SURVEY OF WESTINGHOUSE FACILITIES SUITABLE FOR FUSION HARDWARE DEVELOPMENT AND TEST

FACULTY DESIGNATION	LOCATION	APPLICATION-DATA GENERATED	COMMENTS/FUSION RELEVANCE
Hydraulic Test Facility	W ARD; Waltz Mill, PA	Thermal Striping, Reliability and Performance Data. Two large water loops are used for flow orificing studies, component flow and vibration studies, pump performance studies and component thermal striping. Flowrates: 2000-6000 gpm, at 180°F, 200 psi.	Possible future application for testing water cooled FW/B/S components.
Sodium Materials Test Loops	W ARD; Waltz Mill, PA	Sodium compatibility - chemical, physical and mechanical properties of coatings and other wear-resistant surfaces for LMFBR applications.	Possible future application to fusion reactor components.
Self-Welding System (SWS-1)		Self-welding characteristics of reactor materials in sodium-variable contact periods and loadings.	Possible future application to fusion reactor components. Can be used with a lithium supply system.
Sub-component Test Facility (SCTF)	W ARD; Waltz Mill, PA	Orifice performance, erosion, cavitation and pressure drop studies. Capable of 100 psi/100 gpm flow conditions.	Possible future application to fusion reactor components. Can be adapted for lithium use.
Sodium Friction and Wear Rigs (SWEATER-1, -2, -3)	W ARD; Waltz Mill, PA	Friction and wear behavior of interfaces with sodium testing has also been done with argon, air and helium in support of GCBR.	Possible future application to fusion reactor components. Can be adapted for lithium use.
Thermal Cycling Facility TCF-1, -2, -3	W ARD; Waltz Mill, PA	Tests of materials damage mechanisms and kinetics due to thermally-induced creep and fatigue strains in outer surface regions - conducted under prototype thermal/mechanical and sodium purity conditions.	Possible future application to fusion reactor components, data relevant to many high thermal shock situations.
Sodium Pre-Exposure Loops	W ARD; Waltz Mill, PA	Used to pre-expose mechanical property test specimens to representative sodium conditions	Possible future application to fusion reactor components

TABLE H-1 (CONTINUED)
SURVEY OF WESTINGHOUSE FACILITIES SUITABLE FOR FUSION HARDWARE DEVELOPMENT AND TEST

FACULTY DESIGNATION	LOCATION	APPLICATION-DATA GENERATED	COMMENTS/FUSION RELEVANCE
In-Sodium Mechanical Property Test System	W ARD; Waltz Mill, PA	Creep, creep fatigue, fatigue, simple tension, simple compression, and stress-relaxation data under corrosion conditions corresponding with LMFBR sodium circuits. Tests may be conducted in air, inert gases and other liquid media.	Possible future application to fusion reactor components. Can be used with a lithium supply system.
Thermal Gradient Test Loop	W ARD; Waltz Mill, PA	Sodium and Steam/Water Loops - steam generator duplex tubes under steady state and off-normal conditions.	Possible future application to fusion reactor components.
Mechanical Properties Laboratory	W ARD; Waltz Mill, PA	Broad capability for mechanical property testing of materials for ambient and elevated temperature service. Creep, multiaxial stress, fatigue, low load creep rupture and routine tensile/compression testing.	Available to support all on-site experimental programs.
Chemical Technology Laboratory	W ARD; Waltz Mill, PA	Facilities include inert environment glove boxes with the capability of handling radioactive species. Used for Fission product removal and transport studies, gas phase and aerosol studies.	Available for future fusion R&D-can be used to study tritium behavior purification and removal techniques development and activated corrosion product behavior.
Engineering Test Laboratory	W AESD; Large, PA	Full range of seismic testing, vibration testing and engineering mechanics testing. The seismic facility can handle large plant components weighing up to 40,000 lb. and up to 9 ft. x 9 ft. in size.	Possible future application to fusion reactor components.
Materials Laboratory	W AESD; Large, PA	Facilities include: Materials Science, Metallography and X-ray, mechanical test, thermophysical properties electron diffraction and electron microscopy and chemical vapor deposition. Leader in high temperature, high strength alloy development.	has prepared alloy samples for the CFE ACIP program. Available to support fusion-related R&D.
Hybrid Computer Facility	W AESD; Large, PA	Coupled analog/hybrid computer facility used for complex systems simulation, particularly overall power plant performance studies. SEL 32/77 digital computer with three EAI-2000 analog computers plus peripherals.	Available for fusion R&D. Possible future application to fusion reactor control systems and dynamic systems simulation.

This activity is funded at a relatively low level in FY'81 and the duration of the experimental activity will be approximately two months. commencing in April or May, 1981 depending on the availability of funds for the initiation of work. While it is hoped that this will be an ongoing program. MIT is not in a position to make a long range commitment for facility utilization. The nature of the experimental work involved in the MIT program is quite similar to that proposed for first wall component development so there is a clear opportunity for these programs to share personnel, facilities and information to the benefit of the overall program. Our plan for the sharing of costs for facility operation would consist of allocating the cost of manpower and materials, on an as-expended basis, to each program. When not in use, the facility would be maintained in operational readiness, at Westinghouse expense.

FACILITY DESCRIPTION

The test stand was assembled by coupling an existing high power electron beam generator (developed for E-beam welding and cutting research) to a new pressurized water heat transfer loop and instrumentation and control system. The system consists of the following major components: electron beam gun assembly, beam power supply, cylindrical vacuum tank enclosing the test volume, vacuum pumping system, water heat transfer loop. and a central control and data acquisition system. Figure H-2 is an overall schematic diagram of the system showing the relationship of major subsystems, fluid flow paths and control logic.

The photograph of Figure H-3 labels the major system components which are as follows:

1) 150 kV, 300 mA SF_6 insulated beam power supply (45 kW rating)
2) Oil insulated auxilliary power supplies for beam gun (cathode, bias grid, etc.)
3) 16" diffusion pump
4) Cold trapped 6" diffusion pump

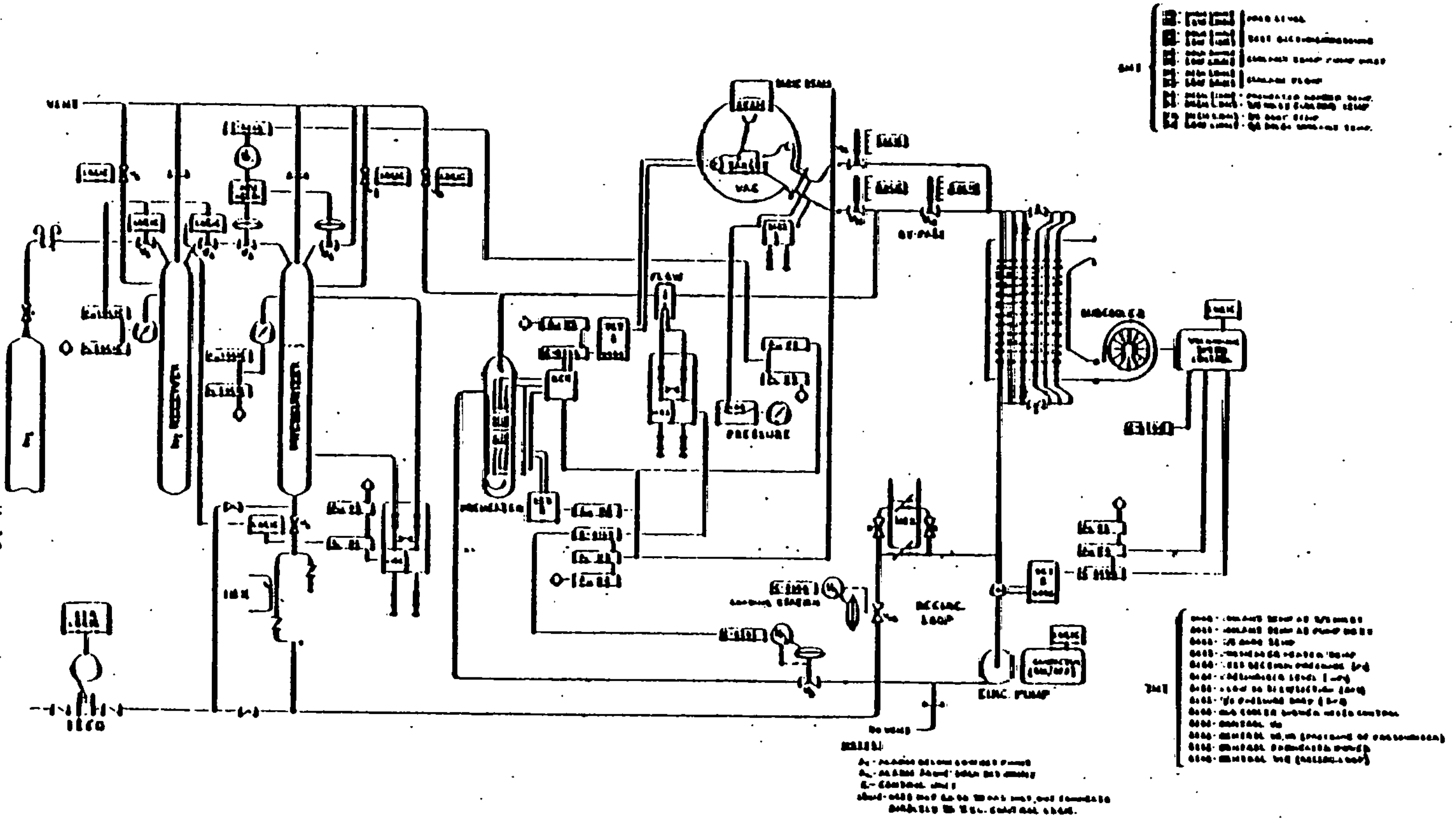


Figure II-2. A Schematic Diagram of the E-Beam/Surface Heat Flux Test Facility Showing Major Coolant Flow Paths and Control Logic

5) Shielded target chamber (1 m diameter x 1.2 m long)
6) E-beam gun assembly

The major components are described in the following subsections.

The Vacuum Chamber

The vacuum chamber is 1 meter in diameter by 1.2 meters long. It can be seen as item 5 in the overall photograph of the test stand in Figure H-3. Ten ports are available for electrical, cooling water and diagnostic feed-throughs and visual observations. The interior of the vacuum chamber is shown in Figure H-4. The various items indicated in the figure correspond to those on the list below:

1) mirror for visual observation of target during operation
2) E-beam sweep coil
3) water cooled beam catcher
4) water cooled slotted mask
5) target mounting base
6) lab water lines for cooling catcher and mask
7) flexible high pressure (1/2") water lines connecting target with heat transfer loop
8) lead shielding
9) vacuum chamber

An aluminum framework inside the chamber allows flexibility in the choice of target orientations underneath the electron beam. The chamber is fitted to a 16 inch diffusion pump (Item 3 in Figure H-3) and a 6 inch trapped diffusion pump (Item 4) which can produce a chamber vacuum of at least 10^{-5} torr. The beam target is an intense source of X-rays; therefore, lead shielding is provid- ed around the target area. A periscope type mirror arrangement with a leaded glass port permits visual observation of the target area during operation.

The E-Beam Gun System

The high power electron beam generating system is illustrated in the schematic drawing of Figure H-5. The electron gun generates a beam of 150 keV electrons at power levels from 1 kW to 36 kW at 240 mA. The beam can be focused on a circular area with a Gaussian intensity profile to a diameter as small as 1 mm, or defocused for a beam diameter as large as 5 cm.

The beam enters the cylindrical vacuum chamber through a top port in which a high frequency magnetic deflection coil is located. The deflection system is capable of sweeping the beam up to 15 cm from the undeflected position at the target location in times as short as 200 μs. The undeflected beam strikes a tungsten block located inside a water cooled copper beam catcher situated next to one end of the target. In present experiments a slotted mask (water cooled copper), located directly above the target, defines the irradiated target area. Both the mask and the beam catcher are electrically floating while the target is grounded. By monitoring the electron currents to the catcher and the mask, total beam power and beam power incident on the sample can be determined. The maximum target area coverage normal to the beam is a function of test bed height within the vacuum chamber. The present test bed height will permit the heating of a 20 cm x 20 cm target. Lowerly the bed will permit the attainment of a normal target area of 30 cm x 30 cm.

A major modification to the beam system which is presently being undertaken is a vast improvement in the scan system. A new scanning coil is being added which allows programmable two-dimensional scans, with a range of $\pm 10^\circ$ in either perpendicular direction. In addition scan rates will double, from 50 cm/ms to 100 cm/ms. Further improvements in the scan system will be possible with the provision of a new coil driver. Faster scan rates and programmed scans would make almost any two dimensional geometry and power distribution realizable.

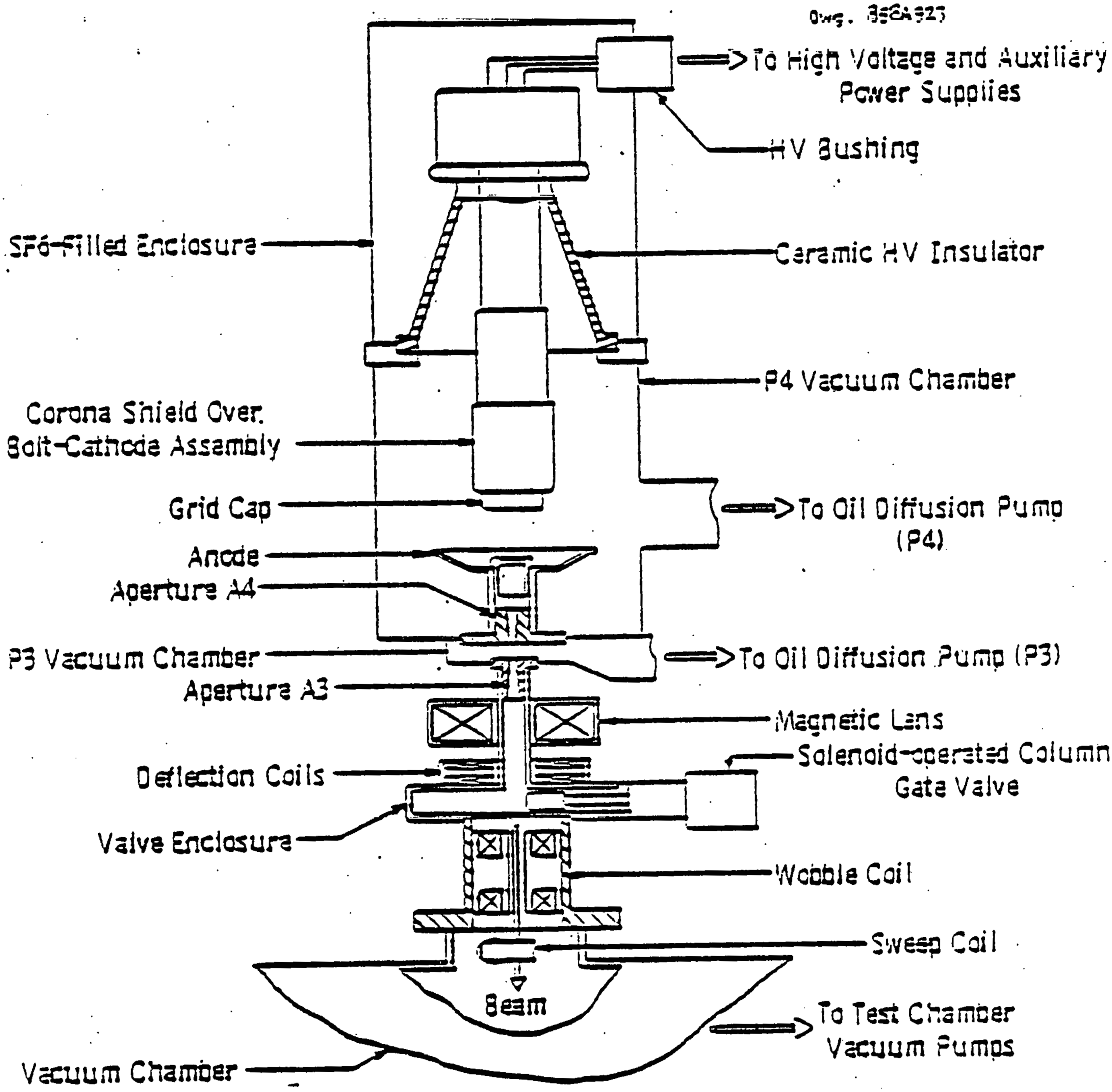


Figure H-5. Electron Optical System and Vacuum System of Test Gun

Pressurized Water Loop

The heat transfer loop used to remove the heat load and to provide the desired heat sink characteristics is shown in Figure H-6. The various items shown in the figure are identified in the table below.

1) variable speed blower for air-cooled subcooler
2) air cooled subcooler (finned tubes inside Al box)
3) flow control valves
4) TI/PM-550 programmable process control system and TTL alarm and interlock system
5) N_2 supply tank
6) 30 kW electric preheater
7) N_2 receiver and pressurizer tanks
8) main circulation pump (centrifugal)
9) pump recirculation loop water cooled heat exchanger

An important consideration in the design of the heat transfer loop was to provide the capability for control of heat sink characteristics in terms of coolant inlet temperature, flowrate and pressure. A 3500 rpm centrifugal pump provides flow of deionized water through the loop, generating flowrates of up to 12 gpm. The maximum working pressure is 400 psia. The system pressure is automatically controlled by a gaseous nitrogen pressurizer. A subcooler consists of a bank of externally finned tubes actively cooled by a variable speed blower enables the removal of up to 200 kW of heat from the water.

Data Acquisition and Control/Instrumentation Systems

The heart of the instrumentation and control system for the loop is a Texas Instruments PM-550 microprocessor based process control system. This is seen as Item 4 in Figure H-6. While continuously monitoring and displaying six selected process variables, the process controller will seek and maintain the system pressure, flow rate, test section inlet temperature, and pump inlet

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temperature as required by preprogrammed instructions. An important additional function of the system is to provide for precise test article calorimetry. The data acquisition system is presently being expanded by the addition of a CAMAC system coupled to a DEC LSI 11/23 microprocessor. The system includes a high speed line printer and information storage discs. This capability permits the acquisition of data during fast transients. This system also serves the light ion beam test stand (see Section H.7) which is now being assembled.

Target instrumentation is also in the process of being modified. An infrared camera system is being acquired for improved target surface temperature measurements. This capability will be operational in early 1981.

APPLICATION AND UPGRADE CONSIDERATIONS

Figure H-7 is a trimetric cutaway view of the test configuration in the electron beam test stand. The test volume available is defined approximately by a cylinder 1 m in diameter by 1.2 m long. This volume is sufficient to permit the installation of a wide variety of small-to-medium scale engineering mock-ups of components with associated mechanical constraints and flow geometries. The height of the test surface within the chamber determines the maximum surface area which can be uniformly scanned to simulate a dc surface energy flux. Test surfaces can be inclined with respect to the beam to increase surface area coverage and test effects related to the angle of beam incidence.

An operating map of the surface heat flux test capability associated with this facility is shown in Figure H-8. With the present beam power capability of 36 kW the surface energy fluxes attainable range from 360 MW/m^2 on a 1 cm^2 section to 0.4 MW/m^2 on a 900 cm^2 section. A relatively inexpensive upgrade has been proposed, utilizing an available power supply and new heat transfer loop components, permitting the attainment of up to 100 kW of beam power on target. In our reference schedule (Section E) we show operation of

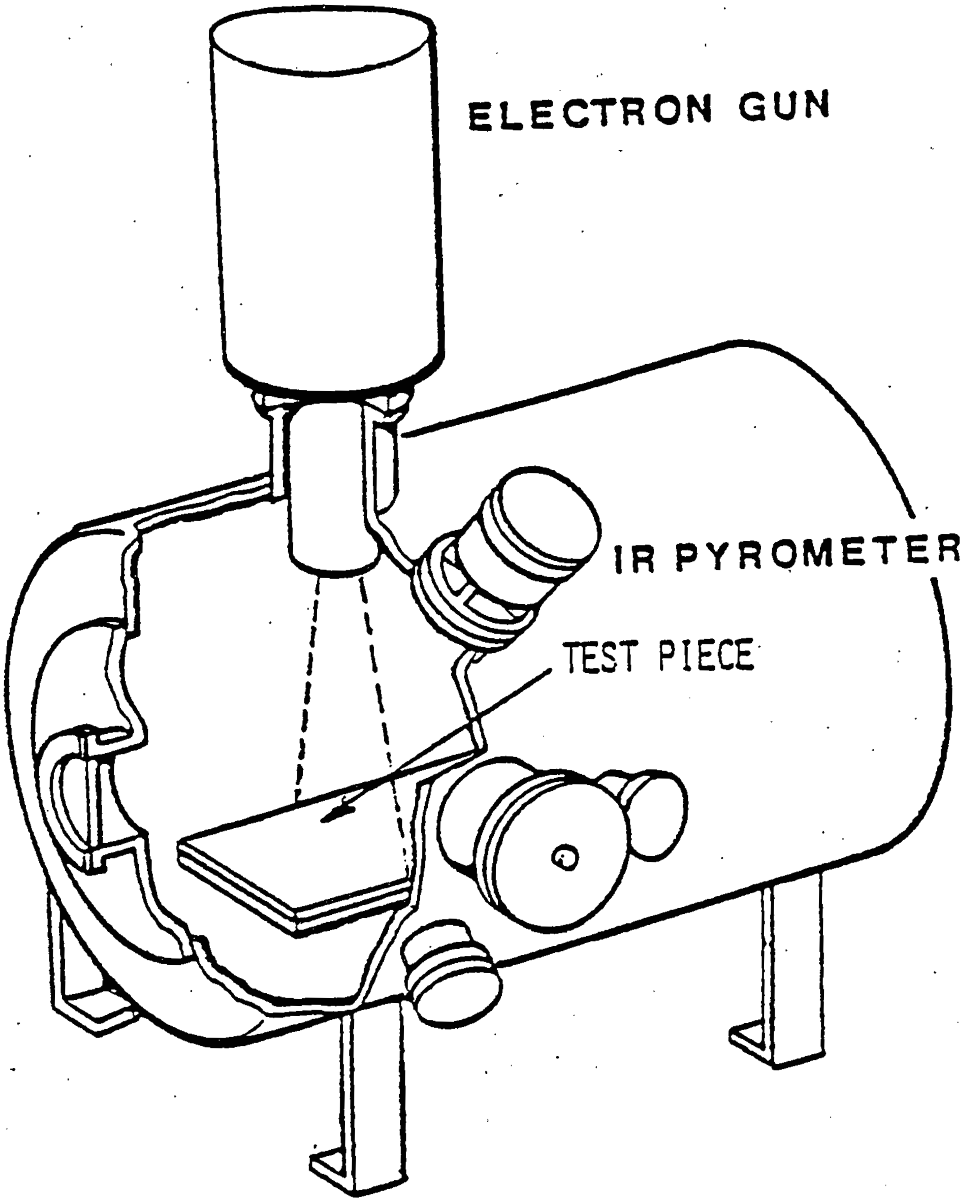

Figure H-7. Schematic Diagram of Neutral Beam Test Stand Illustrating a Test Assembly

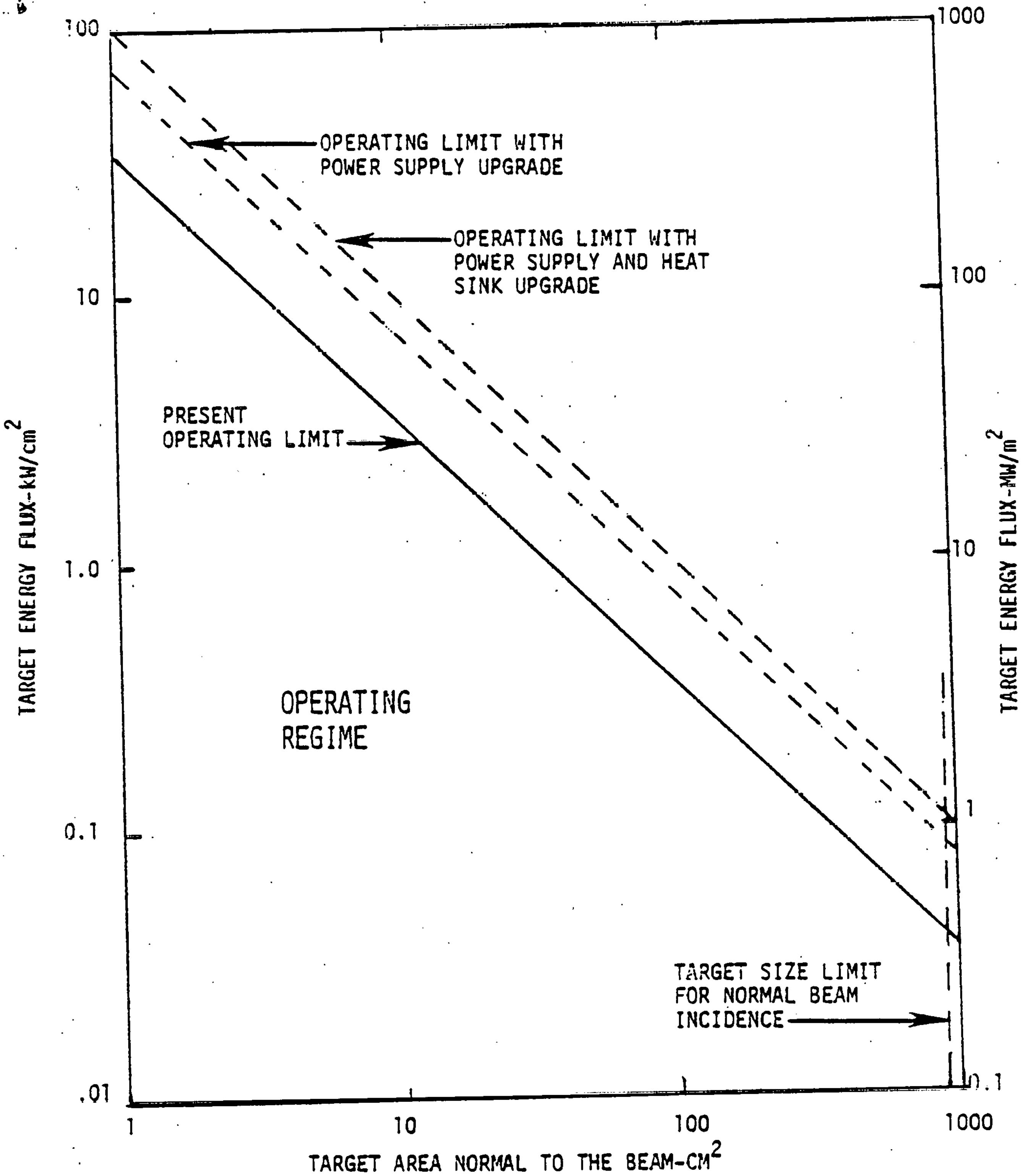

FIGURE H-8. OPERATING MAP OF THE WESTINGHOUSE SURFACE HEAT FLUX TEST STAND (NORMAL BEAM INCIDENCE ONLY) the facility in its present configuration through September of 1981. The upgrade activity would be implemented during the first quarter of FY'82.

The test program which has been recommended in Section C is progressive in terms of test item size and complexity of effects to be simulated. Thus the testing activity during the remainder of FY'81 and FY'82 which has been defined can be fully accommodated by the test stand with the upgrade defined. Subsequent upgrades have been identified; these can be planned and designed in detail in FY'82 and implemented in FY'83. These upgrades include:

- Installation of radiant heat sources to provide a large area (1 m^2) "background" surface heat fluxes in the range of 0.1 to 0.3 MW/m^2.

- Installation of ion source(s) to provide in-situ combined effects testing over areas in the 1 cm^2 to 10 cm^2 range

- Mechanical actuators for low cycle or high cycle imposed mechanical loads

- Installation of a helium heat transfer loop

- Installation of a liquid lithium heat transfer loop. This upgrade is under consideration to support the magnetic divertor technology program where there is some interest in surface heating of lithium wetted surfaces and the heating of liquid lithium droplet curtains (both under vacuum conditions).

The proposed approach for the utilization of the E-beam facility provides for early testing of a technically relevant nature and defers significant hardware expenditures until FY'83. Furthermore, ample time is provided to plan and design some level of combined effects capability which can be implemented within the Phase I time frame and budget profile.

H.2 LARGE SCALE SURFACE HEAT FLUX FACILITY (ASURF)

The second baseline test facility proposed for Phase I surface heat load testing is designed for the testing of full scale stainless steel heat rejection panels and assemblies under moderate-to-low input surface energy flux (<0.5 MW/m^2) conditions. The facility can be implemented on a relatively inexpensive basis through the addition of commercial radiant heaters to an existing large hydraulic loop and heat rejection facility. The rationale for this approach is that the early stages of the test program will concentrate on a large number of relatively small test configurations as the data base is developed and which can be more efficiently handled by the E-beam facility.

As component designs mature it will be possible to construct full-scale mock-ups of designs which have been previously screened and deal with phenomena which cannot readily be scaled down. The issues of interest here are the overall structural integrity and thermal/hydraulics performance of relatively complex flow geometries which are design-specific. Components of this nature which have been designed to date primarily include large stainless steel first-wall heat rejection panels and limiters. In these cases the vacuum environment is not essential and the normal surface loads expected are within the capability of radiant heating sources.

The large water loop is the secondary, heat rejection system of the steam/water loop facility. It is illustrated in Figure H-9. The LWL and other large scale test facilities are located at the Westinghouse Waltz Mill Site (Madison, Pennsylvania). As a secondary loop, its principal function is to transport and reject waste heat through a cooling tower. The system consists of a number of heat exchangers with a total heat rejection capacity of 2 MWt. The capacities of the system components are as follows:

Heat Regenerative Capacity	Condenser	1.3 MWt
	Subcooler	0.5 MWt
Heat Rejection Capacity	Aftercooler	1.3 MWt
	Desuperheater	1.2 MWt
Heat Additive Capacity	Preheater	0.2 MWt
Cooling Tower Capacity		2.0 MWt

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The system has been used primarily as a source of feedwater for LMFBR steam generator test programs. It has been employed in various steam generator weld test programs and steam generator model performance test programs. The facility has successfully logged over 8000 hours in support of these activities.

The basic system is fabricated of 216 stainless steel with a design pressure rating of 3000 psig. The primary pump is a positive displacement, variable speed pump capable of delivering 3800 lb/hr at 2400 psi developed head. The facility is capable of supplying feedwater to sodium-to-water steam generators at temperatures to 605°F.

Data acquisition is via an HP-9600 satellite minicomputer that is connected to an HP-1000 central processor. The system utilizes the HP-2313 real time executive software. Data are recorded on flexible disks utilizing four disk drives. There are currently 500 recording channels with an upgrade capability for 1000 channels.

Present programs utilize a small fraction of the available heat rejection capacity. The proposed facility modification would involve construction of a parallel test circuit and erection of a support frame for test articles. A bank of commercial radiant heaters would be installed in modular panels. Instrumentation, control and data acquisition would be accomplished through the existing system which serves the overall facility. It is intended that final planning and design work for the facility modification be initiated in FY'82 and assembly completed early in FY'83. As in the case of the E-beam facility, operating costs will be principally based on labor and materials which are directly expended on experimental tests.

H.3 PLASMA ARC HEATER FACILITY

Westinghouse has available, at its East Pittsburgh facility, a laboratory devoted to the development and application testing of plasma arc heaters. The facility is located next to the Westinghouse High Voltage Laboratory which can provide up to 3 MW of electrical power to the arc heater laboratory on a steady state or transient basis. The laboratory includes a central control and data acquisition system, a variety of process gas supply systems (including hydrogen, helium, and argon), a water heat rejection loop and a sodium loop for both process and heat rejection purposes.

While a specific test program involving arc heaters for first-wall surface testing has not been identified in this response, a description of our development facility and recent applications is provided to indicate our awareness of this technology and to indicate the availability of the facility should an appropriate application opportunity arise during Phase I. Westinghouse is a commercial supplier of a line of plasma arc heaters which are available in several sizes and ratings, with individual unit power ratings up to 3500 kW. The test facility at East Pittsburgh currently has two models available for development testing: the Mark II and Mark 31 models. The following describes recent test applications involving these units.

The Mark II unit was used in a magnetite ore spheroidization program in the configuration shown in Figure H-10. The heater is a self-stabilizing device with 1 MW rating. The figure shows the arc heater assembly and the heating target chamber. The cylindrical chamber has an inside diameter of 20.3 cm. Heating chambers of different lengths can be accommodated. The arc heater efficiency is typically 70%; the efficiency varies slightly, depending on the gas used. Gases are heated to a temperature $\leq$9000 K at the electrodes and 3000 to 5000 K in the test chamber. Normal target areas up to 100 cm^2 can be accommodated with surface heat loads in the range of 35 to 50 MW/m^2 at a distance of 4 inches from the electrodes.

The Mark 31 unit was designed as an ultra-high temperature heat exchanger for industrial plasma chemistry applications. Individual units are rated at power levels up to 3500 kW and are designed to operate on standard 60 cycle AC power systems. A recent application of these heaters was in an experimental solar grade silicon process unit developed and tested for the Jet Propulsion Labora-

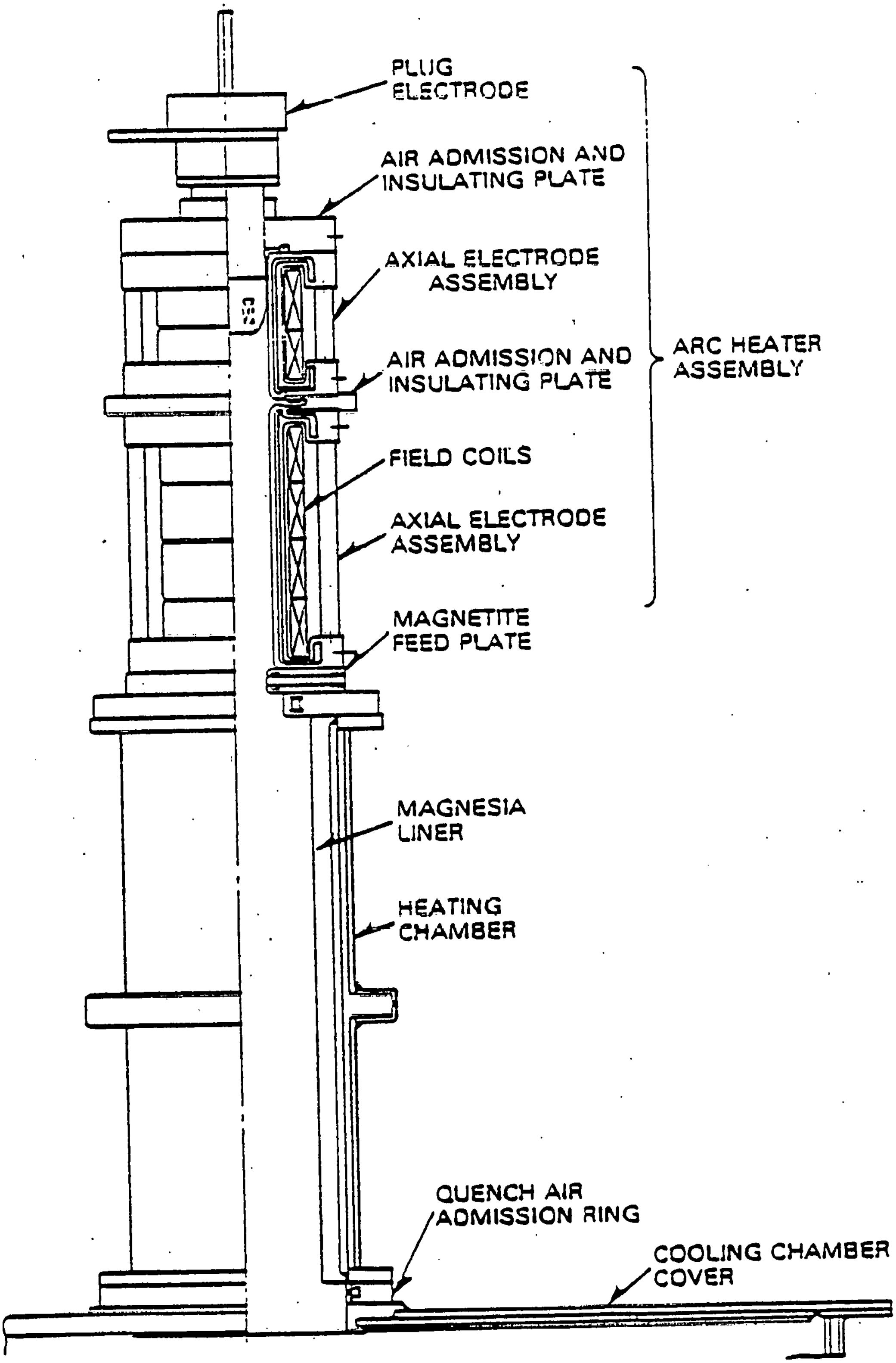

Figure H-10. Section View of the Westinghouse Mark II Arc Heater Assembly Used in a Magnetic Ore Spheroidization Program tory. The system, shown in Figure H-11 uses three 500 kW Mark 31 arc heaters connected to a chemical reaction chamber. The process gas is a hydrogen-argon mixture. Figure H-12 is a cut-away of the reaction chamber. In operation, liquid sodium is sprayed through the arc heater gas stream. The high temperature argon-hydrogen gas causes vaporization of the sodium droplets for subsequent reaction with silicon tetrachloride.

H.4 GENERAL PURPOSE BOILING HEAT TRANSFER LOOP

A general purpose boiling heat transfer loop is located at the Westinghouse Research and Development Center, in close proximity to the electron beam test facility. A specific role has not been identified for this facility in the Phase I FW/B/S program for TPE I; however, a description of this facility is provided in the event that additional off-line testing capacity is required for boiling and two-phase flow studies where precise modeling of surface heat flux is not required. The loop may be operated with R11, R114, hydrocarbons, or water as the working fluid. A particularly relevant application of the apparatus has been detailed studies of cross-flow boiling heat transfer simulating nuclear steam generator conditions.

The hydraulic circuit consists of a canned motor pump, a test loop, a by-pass loop, a decouplable pressurizing limb, a preheater, and a heat exchanger for heat rejection. The pump is capable of circulating 2×10^4 lb/hr of water at 128 psia and 250°F. The loop piping is made of stainless steel and is rated at 1000 psig. Primary control of the test flow loop is by a pneumatically actuated flow control valve. A by-pass loop permits the throttling down of the test loop flow to about 5% of the maximum flow.

A 21 kW preheater is used to control fluid temperature or vapor quality at the entrance to the test section. The heat output rates of the preconditioning section and the measuring section are separately adjustable so that the entering condition to the measuring section is independent of the surface heat flux to the section. The loop pressurizer is of the gas over liquid type. Nitro-

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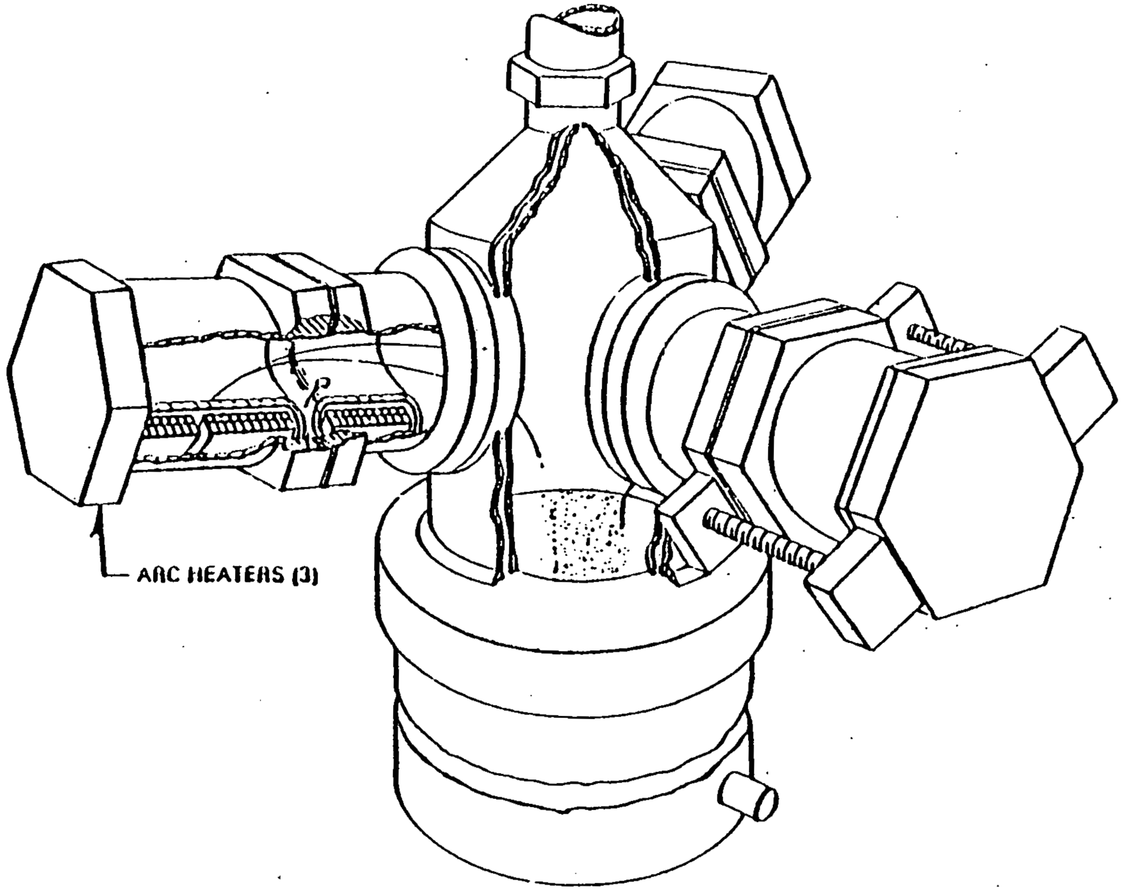

Figure II-12. Schematic Diagram of the Plasma Arc Heater and Reaction Chamber Configuration in the Experiment Silicon Production Process Line gen gas can be admitted to the pressurized vessel via a fine control valve to boost the pressure.

H.5 GENERAL PURPOSE SODIUM LOOP-1 (GPL-1)

The GPL-1 loop is shown in Figure H-13. It is located adjacent to the Large Water Loop described in H.2. The primary piping system is 2-inch Schd 40, designed to operate to a maximum sodium temperature of 1200°F at a maximum design pressure of 330 psig. During its operational history, this facility has been employed in the following tests:

a) Fuel-Pin Heat Transfer tests
b) Driver Fuel Assembly Wear and Friction Tests
c) Subassembly Fretting and Wear Test
d) Fuel Assembly Nozzle Leakage Tests
e) Westinghouse Oxygen and Hydrogen On-Line Metering Tests
f) Westinghouse Ultrasonic Flowmeter Tests
g) Radial Blanket Heat Transfer Tests

In conjunction with the Westinghouse High-Temperature Steam Water Loop, the GPL-1 facility has been employed in the following tests:

a) Aluminum-Bonded Steam Generator Evaluation
b) TST-1 Steam Generator Tube-to-Tube Sheets Weld Test Evaluation
c) TST-2 Steam Generator Tube-to-Tube Sheets Weld Test Evaluation
d) HTM-1, Megawatt Steam Generator Evaluation
e) SSGM. Duplex-Tube, Steam Generator Evaluation

In execution of the TST-1 and TST-2 test programs, the GPL-1 facility subjected the test articles to a total of 1000 thermal shocks of 300°F in 10 seconds. In execution of the HTM-1 steam generator evaluation tests over 5000 hours of high temperature operation was carried out in (1) heat transfer evaluation, (2) fluid flow dynamics, and (3) off-chemistry materials evaluation. The test

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silo has been used in the steam generator tests and is available for future steam generator evaluation.

Heat transfer tests on core and blanket fuel assemblies can be performed using an electrically heated bundle assmbly installed in General Purpose Loop No. 1. The facility is currently being used on a program to characterize heat transfer performance of a prototype, full sized 61-rod radial blanket assembly.

H.6 HYDRAULIC TEST FACILITIES AT THE WALTZ MILL SITE

The Hydraulic Test Facility consists of two large water loops designed to act either independently of each other, or in conjunction if specific programs require such operation. These two loops are the Thermal Mixing Hydraulic Loop (TMHL) and the Multi-Purpose Hydraulic Loop (MPHL). The former is a six-inch piping system with flow capability to 2000 GPM at temperatures to 180°F; the latter is a twelve-inch piping system with flow capability to 6000 GPM at temperatures to 180°F. Complete instrumentation, controls, and Data Acquisition Systems are available. A schematic diagram of MPHL is shown in Figure H-14.

The two loops also employ the "parallel test section" concept, and have been used on a wide variety of CRBRP, LMFBR, and PWR tasks over the past twelve years. Most recently, their application has included tasks in support of Radial Blanket Flow Orificing, Steam Generator Flow Modeling, vibration analysis of PWR heat exchangers, and flow distribution analysis of reactor internals of a PWR reactor. Most importantly, however, has been the program effort on the study of Thermal Striping phenomena in the upper internals region of the CRBRP primary vessel. These two facilities have proven to be exceptionally useful in evaluating thermal striping behavior; and they together represent an operational facility ready to extend the experimental scope and knowledge of this critical phenomena.

Through either concurrent or independent operation, the two facilities provide a "cost-sharing" option to minimize the operating costs of multiple programs

P = PRESSURE GAGE
T = TEMPERATURE INDICATOR
ΔP = DIFFERENTIAL PRESSURE GAGE
S = STRAIN GAGE
SB = SWITCH AND BALANCE UNIT
SI = STRAIN INDICATOR
HV = HAND VALVE
RCV = REMOTELY CONTROLLED VALVE

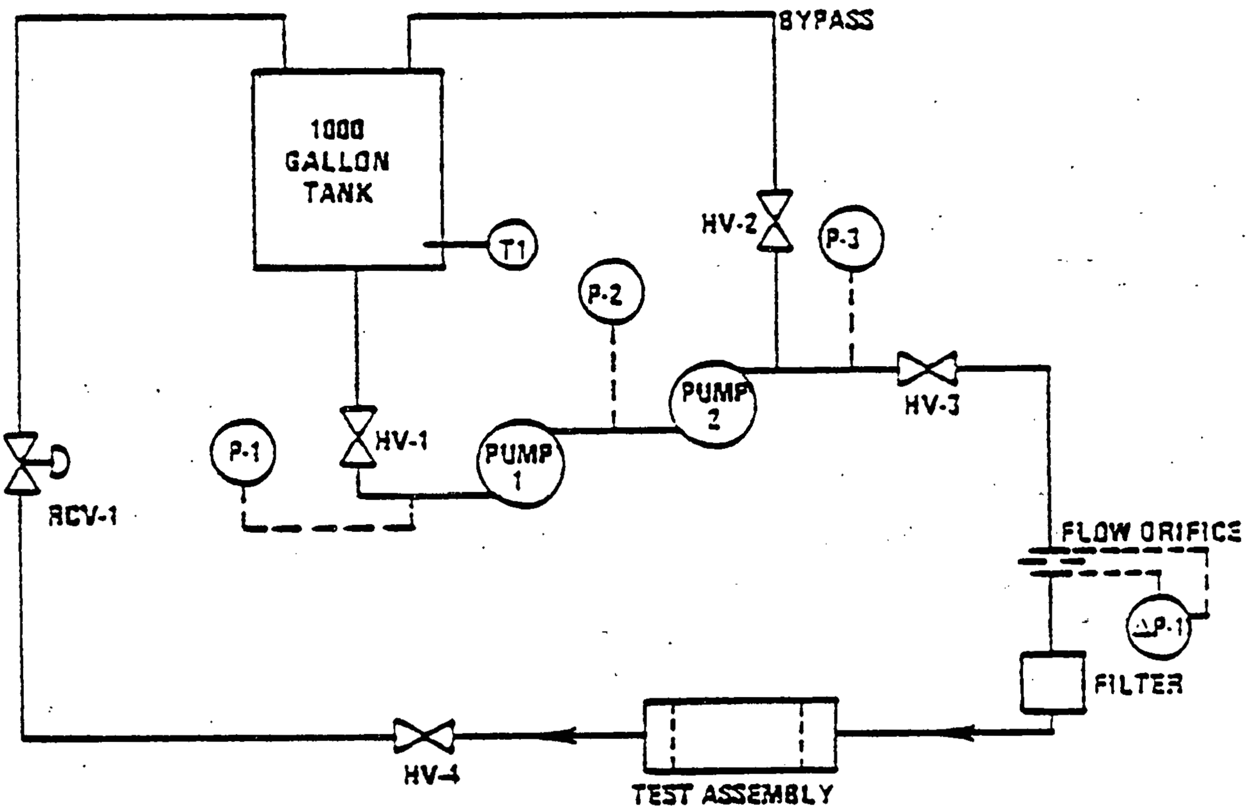


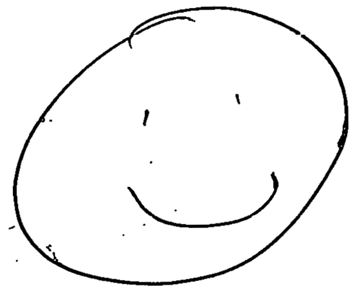
Figure H-14. Multi-Purpose Hydraulic Loop (MPHL) at the Westinghouse Waltz Mill Site (Madison, PA) being carried-out concurrently in the Hydraulic Facility. The ease of startup and shutdown permits one or two-shift operation on most programs. The operating and design parameters of the two loops are as follows:

	TMHL	MPHL
Maximum Design Temperature	180°F	180°F
Maximum Design Pressure	200 psi	200 psi
Design Basis	Power Piping Code ANSB 31.1.0	Power Piping Code ANSB 31.1.0
System Fluid	Water	Water
Maximum Flow Rate	2000 gpm	6000 gpm
Pump Head	100 psig	100 psig
Heat Dump Capability	200 kW	200 kW
System Capacity	3400 gal	1500 gal
Filtration	Full Flow Filter	Full Flow Filter
Piping (Type 304 SS)		
• Main System	6 inch	12 inch
• Auxiliary	3 inch	1.4 and 6 inch
Flow Measurement		
• Main Stream	6 inch Orifice	12 inch Orifice
• Auxiliary	3 inch Orifice	1 and 4 inch Flowmeters 6 inch Orifice

This facility is available for fusion-related programs in the areas of (1) pressure drop, (2) flow-induced vibration, and (3) flow distribution in reactor components and assemblies.

H.7 LIGHT ION-BEAM SURFACE EFFECTS FACILITY

A light ion beam test stand is in the final checkout stage at the Westinghouse Research and Development Center, in the same laboratory housing the E-beam surface heating test facility. Light ion beam energies in the range of



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1 to 150 keV will be available at a maximum power (at 150 keV) of 1.5 kW. Energy fluxes of ~5 kW/cm^2 will be attainable for beam areas of ~0.3 cm^2). This facility will be initially devoted to Westinghouse supported surface physics studies, but will be available for DOE supported activities and in particular, can be used to initiate limited combined effects testing in the context of Phase I of the FW/B/S program. The apparatus has been designed for eventual interfacing with the E-beam facility test chamber (through an available port) to permit simultaneous ion beam and electron beam bombardment.

A photograph of the partially completed facility is shown in Figure H 15. The facility contains the following components:

- A Siemens duoplasmatron ion source capable of light ion currents ranging to 10 mA (pulse and CW operation);

- An extraction-focus-acceleration system capable of operation over a range extending to 150 kV;

- A bending magnet for mass-energy selection and beam switching;

- A differentially pumped (Zr-Al) scattering chamber and associated detection instrumentation (i.e., backscatter detector, residual gas analyzer, etc.);

- A direct imaging ion probe mass spectrometer for off-line target analysis (described in Section H.9)

- A DEC LSI 11/23 microprocessor based data acquisition and control system interfaced via CAMAC to the instrumentation and ion accelerator. This system is shared with the electron beam test facility and is designed to operate in a manner similar to the system now operational on the Princeton PLT neutral beam test stand.

Figure H-16 shows the Duoplasmatron and Einzel lens assembly during the assembly process.

H.8 THE COFFEE CO_2 LASER SYSTEM

Westinghouse has available at its Research and Development Center a number of experimental laser systems which could be used for specialized surface heating studies. A high power CO_2 laser system is available and is based on the use of a Continuously Operating Fast Flow Electrically Excited (COFFEE) laser concept. The COFFEE laser uses a high pressure (0.1 to 1 Atm) direct current self-sustained glow discharge excitation scheme which is well-suited for a continuous wave CO_2 laser. A schematic diagram of the COFFEE CO_2 laser is shown in Figure H-17. A photograph of the laser assembly is shown in Figure H-18.

The laser runs continuously but it also can be operated in a pulsed mode. The beam source is one centimeter in diameter. It can be focused on a target as small as 1 mm in diameter, delivering a surface heat load of 640 MW/m^2, or it can be defocused. At 1 cm in diameter, the peak surface heat load attainable is 38 MW/m^2. The system can be upgraded to a beam power capability of 25 kW.

H.9 STRUCTURAL BEHAVIOR OF MATERIALS

As a leading developer and manufacturer of components and structures for nuclear, underseas, aerospace and defense applications, Westinghouse has developed a unique concentration of capabilities at its Research and Development Center to analyze and characterize the response of structural materials to a wide variety of conditions associated with these advanced engineering applications. The expertise and facilities in this area are an important part of the total capability that Westinghouse offers to support ANL in the performance of the FW/B/S Engineering Test Program.

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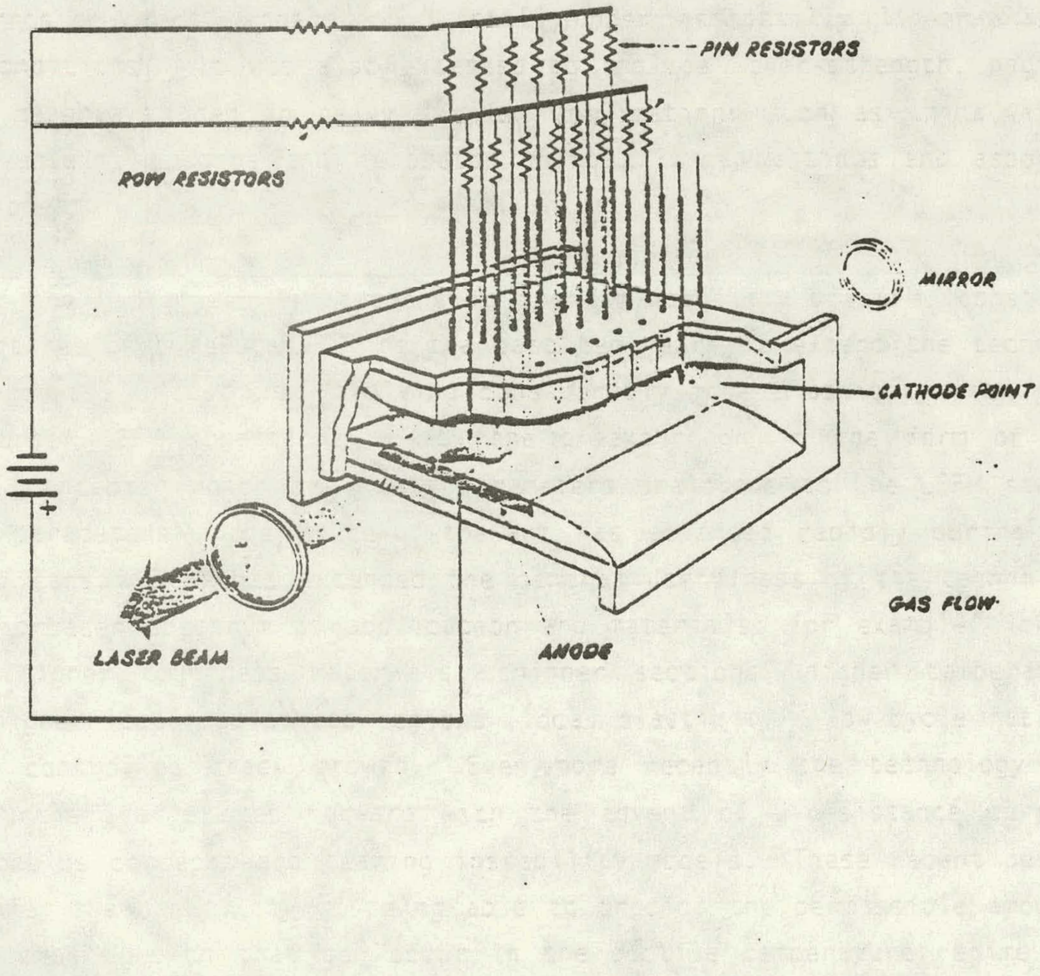


Figure H-17. A Schematic Diagram of the COFFEE CO_2 Laser System at the Westinghouse Research and Development Center.

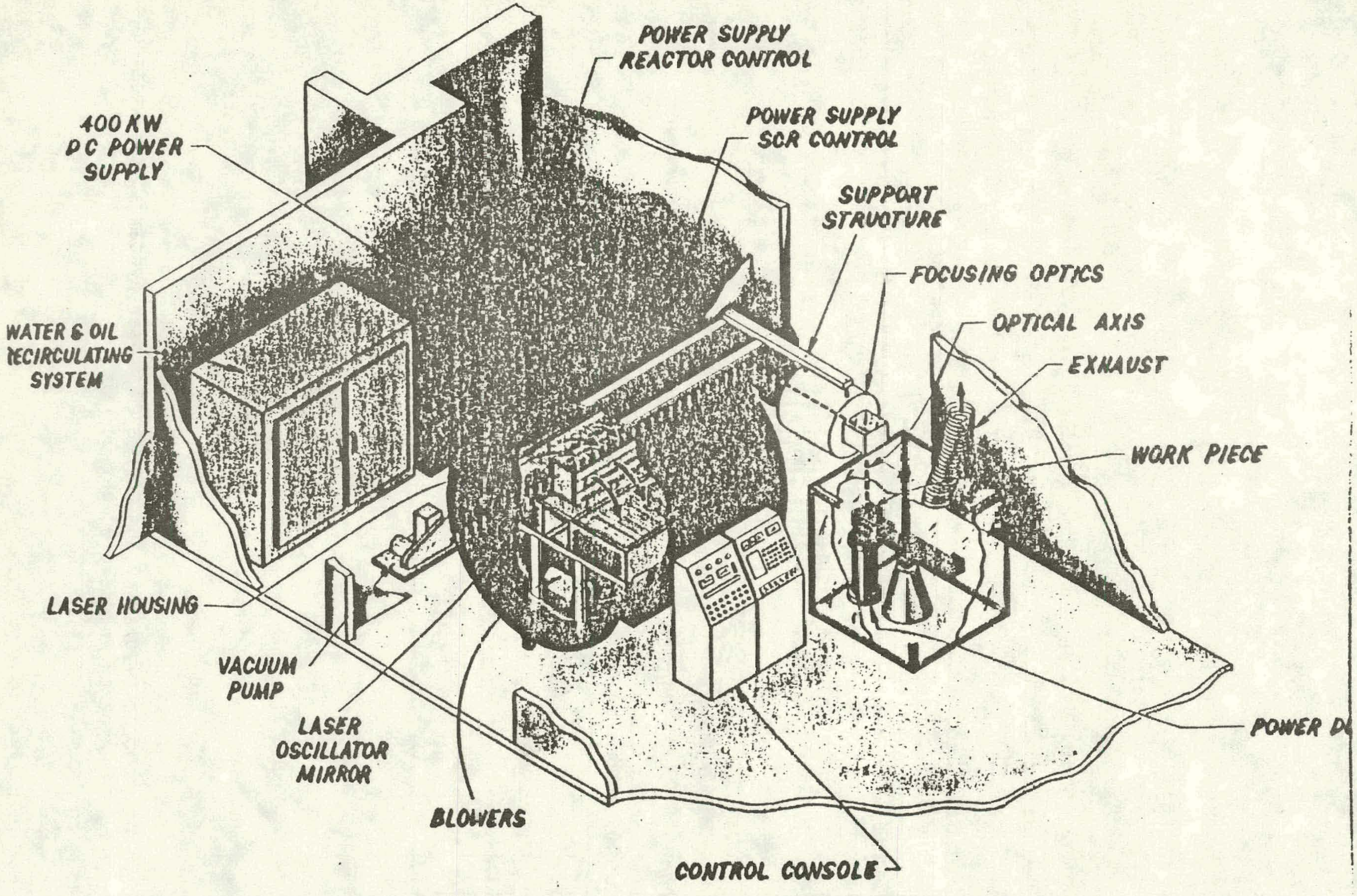


Figure H-18. Compact COFFEE Laser Facility for 25 kW Scaling Tests

STRUCTURAL ANALYSIS

Westinghouse personnel have pioneered in the application of finite element techniques for structural, thermal/hydraulic and magnetic analysis. This includes the antecedent work for the development of ANSYS during the NERVA (nuclear rocket) program and the subsequent development of WECAN, a Westinghouse Proprietary system of codes. The applications of WECAN include the following:

- Static Analysis - Stresses and deformations in structures due to thermal and mechanical steady-state loading conditions. The material behavior can be nonlinear (thermoplasticity, creep, friction and contact).

- Dynamic Analysis - Dynamic response of structures due to steady-state and transient loadings; natural frequency and mode shape determinations of structures; and seismic time history analyses and response spectrum calculations of structural systems such as piping systems, electrical equipment, etc. The modal superposition method of time history analysis is also available.

- Heat Conduction - The steady-state or transient temperatures in structures can be determined so that they can be used to compute thermal stresses.

The WECAN element library consists of sixty-five elements that can be used to model beams, shell structures, two- and three-dimensional solids, springs, dashpots, etc. Special crack-tip elements have also been developed. In addition to WECAN, various commercially available programs, such as NASTRAN and ANSYS, are also available.

FATIGUE AND FRACTURE MECHANICS

The Westinghouse R&D Center has done significant research and engineering work in fracture mechanics and fatigue for many years. The early developments and applications focused on linear-elastic or plane-strain fracture mechanics (LEFM). The original use of the LEFM technology was limited to fracture under essentially linear-elastic loading conditions and relatively high-strength brittle materials such as aircraft structures, missile cases, gun tubes, etc. Subsequently the technology was extended to encompass fatigue crack and stress-corrosion crack propagation - still under essentially linear-elastic loading conditions. It was also extended to include lower-strength, higher-toughness materials used in heavy section applications such as thick walled pressure vessels, turbine and generator shafts, large castings and assorted heavy equipment.

Because of the recognized limitations in the applicability of LEFM, considerable effort has been devoted during the past ten years to extend the technology to encompass situations involving considerably more plasticity than permissible under LEFM conditions. With the breakthrough in the form of the J-integral concepts (which are field parameters analogous to the LEFM stress intensity parameters), the state-of-the-art has advanced rapidly during the past five years. This has extended the general usefulness of the technology to a much broader spectrum of application and materials; for example, lower-strength, higher toughness materials, thinner sections, higher temperature regimes, higher localized stress regions (local plasticity), low cycle fatigue and creep controlled crack growth. Even more recently the technology has taken a further large step forward with the advent of J-resistance curves, tearing modulus concepts and tearing instability models. These recent developments offer the capability of being able to predict the permissible amounts of stable crack growth that can occur in the ductile temperature regime and the eventual instability conditions for the catastrophic failure of the structure by ductile tearing under fully plastic loading conditions. Even more importantly these recent advances in technology offer the promise of being able to design structures and select materials to avoid the possibility of ever being capable of failing due to ductile tearing instability. Thus, the potential failure mode could be limited to plastic collapse or limit load failure criteria.

Fatigue and plastic deformation at stress raisers has been studied in some detail, both experimentally and analytically, including work on finite element modeling. Specific experimental and analytical studies have been made of the problem of a small crack growing from a stress raiser under fatigue loading. Plasticity effects on crack growth in fatigue have been studied from the viewpoint of fracture mechanics, with the J-integral being first used at Westinghouse R&D for such study.

Concerning time dependent effects, crack growth due to creep strain has been studied from the fracture mechanics viewpoint. A modified time-based J-integral approach to this problem was originated at Westinghouse. In-house work is underway in this area which is related to nuclear and steam turbine applications, with this work including study of combined fatigue and creep loading. Analytical studies using finite elements have been made of cracked bodies under plastic and creep loading, and such work with in-house funding is underway at the present time.

Westinghouse R&D Center has extensive experience in low cycle fatigue and fracture mechanics testing of all types. Personnel in the Structural Behavior of Materials Department have contributed extensively to the development of various ASTM test standards, through ASTM task group membership and round robin testing, including ASTM E606, Recommended Practice for Constant-Amplitude Low-Cycle Fatigue Testing. and ASTM E647. Tentative Test Method for Constant-Load-Amplitude Fatigue Crack Growth Rates Above 10^{-8} m/cycle

EXPERIMENTAL STRUCTURAL MECHANICS

The present facilities for mechanical testing of materials and components in the Structural Behavior of Materials Department at Westinghouse R&D Center include ten electrohydraulic test systems which draw their hydraulic power from two 70 gpm pumps. The test systems include three 15 kip load frames (in-house construction) utilizing MTS controls. One of these units is presently being used for testing up to 3500 psig $H_2 + H_2S$ at 850°F. Additional electrohydraulic systems are a 25 kip MTS system, a high rate 20 kip MTS system, two high frequency (200 Hz) test systems - 10 kip and 20 kip. a 100 kip MTS system and a 500 kip MTS test system. Dependent only on specimen size and geometry, temperature testing capabilities range from -452°F to +1200°F. Environmental testing capabilities include all aspects of fracture toughness testing in aggressive environments such as steam, H_2, and H_2S and solutions such as boric acid and sea-water.

A Westinghouse 2500 computer is presently being used for data acquisition and reduction. An MTS PDP 11/34 computer system with 64 K of storage is used for single test machine control, and also for multiple test data acquisition and reduction, with expansion to multiple test control underway. Several microprocessors are also in use for test control.

Supporting equipment in the lab facilities at this time include two 20,000 lb tensile machines (in-house construction), a 50,000 lb. Baldwin universal mechanical test machine. a 240 ft-lb Charpy impact machine, and creep testing machines. Also available are a wide range of appropriate test chambers, instrumentation, controllers, and calibration devices and also equipment for ultrasonic and electric potential monitoring of crack growth. Additional test equipment and apparatus are available from other groups in the Westinghouse R&D Center.

MICROSTRUCTURAL EVALUATION

An important aspect of the proposed experimental approach for the thermo-mechanical testing of FW/B/S components is the pre-test and post-test characterization of the condition of test articles. This characterization can be accomplished as well during test sequences. The facilities available at the Westinghouse Research and Development Center provide access, as required, to the most up to date techniques for the destructive and non-destructive evaluation of materials at the macroscopic and microscopic levels. Recent facility improvements have provided for improvements in resolution accuracy and turn-around time through the use of computer controlled equipment.

Facilities equipment and techniques available include:

- CAMECA SMI-300 Direct Imaging Mass Analyzer (DIMA) has been in operation since 1973. Experience with both spark source and ion probe mass spectrometry provides a unique capability, in that the combination of these two techniques is indispensable for the complete characterization of materials. The CAMECA SMI-300 instrument is equipped with an AS-200 multichannel analyzer with 200 channels for storage. It is primarily used to obtain concentration profiles as a function of depth, and can monitor two elements or isotopes at a time.

- Electron Beam Microanalyzer (MAC-400) - Comprised of three wavelength-dispersive X-ray spectrometers for qualitative and quantitative analysis with backscatter and adsorbed electron measuring facilities for point analyses and line profile recording at resolutions better than 0.5 micron.

- Scanning Electron Microscope, Cambridge Stereoscan Mk.IIa with energy dispersive X-ray spectroscopy attachment.

- Auger Electron Spectroscopy (AES) analyzes the first 3 to 5 atomic layers of an exposed surface for all elements except H and He. The sensitivity for heavier elements is comparable to that of the electron microprobe, but AES is more sensitive to the lighter elements. Typical detection limits are of the order of 0.1 atomic percent (1000 ppm) fairly constant across the periodic table. The technique is most valuable in combination with layer removal methods (sputtering) where the extreme depth resolution of AES (10 Å) is limited only by sputtering damage considerations and sputtering rates. Auger Electron Spectroscopy, ESCA (XPS X-ray Photoelectron Spectroscopy) SAM (Scanning Auger Microscopy) and MACS (Multiple-Technique-Analytical Computer System) are all integrated into a PHI-548 Surface Analysis Unit.

- Scanning Transmission Electron Microscope, JSEM-200C - Capable of imaging precipitates and defects with 1-2 Å resolution: can do chemical analysis of localized regions with a 200 Å resolution; can determine distribution of deformation by electron channeling with a resolution of 200 Å.

- The Rutherford Backscattering System (RBS) is capable of profiling impurities and defect concentrations nondestructively in semiconductors.

- Crystallography and Electron Diffraction
- Hardness Measurements
- Ion Milling
- Light Microscopy and Photomicrography
- Particle Size Analysis
- Phase Contrast Microscopy
- Residual Stress Analysis
- Ultraviolet Photoelectron Spectroscopy
- X-ray Diffraction
- X-ray Photoelectron Spectroscopy

The services of a Nondestructive Test Development Section are available to aid in monitoring growth of small cracks in this program. They have capabilities and equipment in the areas of ultrasonics, eddy currents, radiography, magnetic techniques and acoustic emission.

H.10 VACUUM ARC RESEARCH

As the size of tokamak devices increases, there is increasing concern over possible surface damage due to arcing which has been observed to occur across the boundary layer or sheath at a plasma-metallic wall interface. Professor Miley has estimated that this interaction could lead to erosion rates several orders of magnitude larger than for normal ion sputtering.

Westinghouse offers to conduct a theoretical and experimental program related to unipolar arc phenomena and the evaluation of engineering solutions to the problem under the auspices of the ANL FW/B/S Program. The program would be conducted by scientists and engineers in the Power Interruption and Lamp Technology Department at the Westinghouse Research and Development Center.

Unipolar arcs have been generated and studied in a prior experimental study at Westinghouse. This work could be extended to include the following areas of investigation:

- Measurement of ion fluxes, particle fluxes and erosion associated with cathode spots on typical first wall component surfaces.
- Study of the cathode spot erosion rate under the influence of applied magnetic fields (typically parallel to the surface).
- An evaluation of arc mitigation techniques. This will involve experimental evaluation of materials, surface finishes and coatings.

Recent Westinghouse research programs have concentrated on developing experimental facilities for research into high voltage and high current circuit interruption; these facilities are well suited for vacuum arc phenomena research. An important hardware activity just completed is the development and test of the Ohmic Heating Interrupter for TFTR. This unit is rated for opening a 25 kA current and then withstanding a voltage of 25 kV which rises to that value in about 500 microseconds. Development and test facilities include (1) a high power capacitor bank which incorporates a high voltage synthetic circuit to simulate high voltage and high current transmission and distribution systems; (2) two demountable vacuum systems for the study of vacuum arc phenomena; and (3) the apparatus and diagnostic methods to investigate vacuum breakdown at very high voltages. The special techniques and diagnostic capabilities developed with these facilities are available for use in the proposed program, enabling the work to be performed without major additional test equipment and with a corresponding savings in time and money.

The basic test facility power supply consists of a capacitor bank capable of storing a maximum energy of 2×10^6 joules, several air core inductors, and appropriate instrumentation. The bank produces an oscillatory current when discharged through the air core reactors and the test device. The primary purpose of the test facility is to provide high power for short periods of time. The capacitor bank consists of 864 capacitors (50 µF), each capable of being charged to 10 kV. The bank is assembled such that the individual capacitors form four units, each unit having a maximum capacitance of 10,800 µF. As a consequence, the bank can store the maximum available energy in three distinct modes for operation at: 10. 20. and 40 kV. A 30 kV operating mode is also available by connecting three of the 10,800 µF units in series. In this mode the total available energy is 1.5×10^6 joules.

Figure H 19 depicts the basic circuit of the test facility. The capacitor bank is charged to a preselected voltage (0 to 40 kV) by a constant current power supply which automatically disconnects from the bank at the desired voltage. A control-timer is then manually activated and all subsequent circuit operations are automatically programmed. This circuit will be employed primarily for arcing studies on test items which are located in either one of two demountable vacuum chambers.

HIGH CURRENT ARC CHAMBER

A demountable vacuum chamber is available for performing arcing behavior studies on electrode configurations. This facility, which is shown schematically in Figure H-20, is being used with the 2 MJ capacitor bank for high current testing and evaluation of electrode designs. The entire demountable chamber is fabricated from non-magnetic stainless steel in order to facilitate the use of magnetic fields. The main test chamber is designed so that rubber seals may be used for quick turn-around experiments requiring only modest vacuum. The test chamber can be used with metal seals. however, if bakeout is required for high vacuum work. Large side ports permit direct observation of arc behavior as well as the insertion of arc shields, auxiliary electrodes, and magnetic field coils. The primary test chamber is 25 cm in diameter and 22.5 cm high. The pumping arm and observation ports are 15 cm in diameter. A ceramic isolation insulator permits the chamber to be operated ungrounded during testing. The anode and cathode electrodes are insulated from the chamber using 8 in ceramic insulator sections from a commercial Westinghouse vacuum interrupter.

The demountable arc chamber is presently being used to photograph high current 50 kA vacuum arcs using high speed movie and streak photography. Diagnostics include measurements of arc current and voltage, recovery behavior, and magnetic fields.

LOW CURRENT DEMOUNTABLE SYSTEM

Arcing studies at relatively low currents to 5 kA will be conducted in the demountable chamber shown in Figure H-21. In this chamber the arc plasma is bounded by a 27-cm-diameter metal wall which is isolated from both electrodes

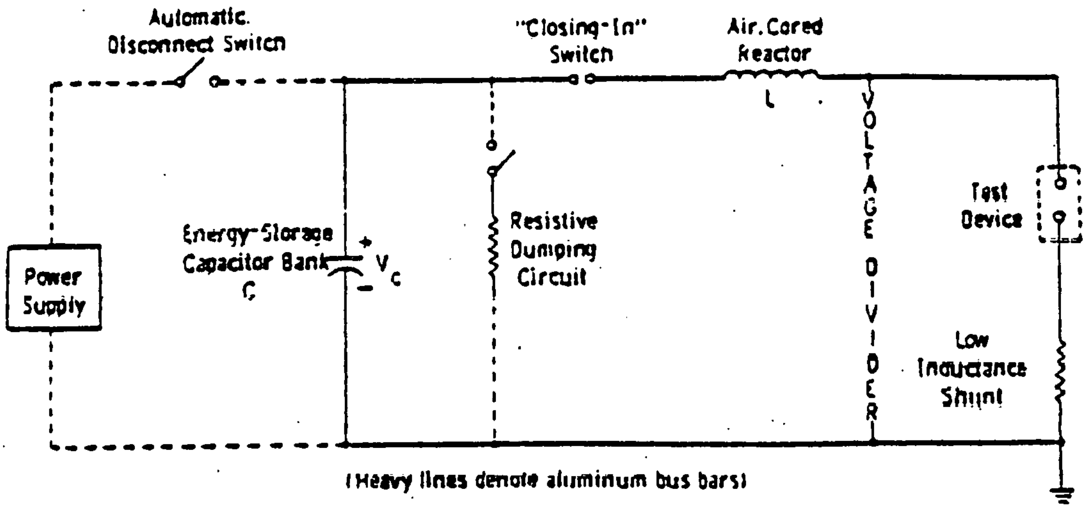


Figure H-19. Vacuum Arc Test Facility Circuit

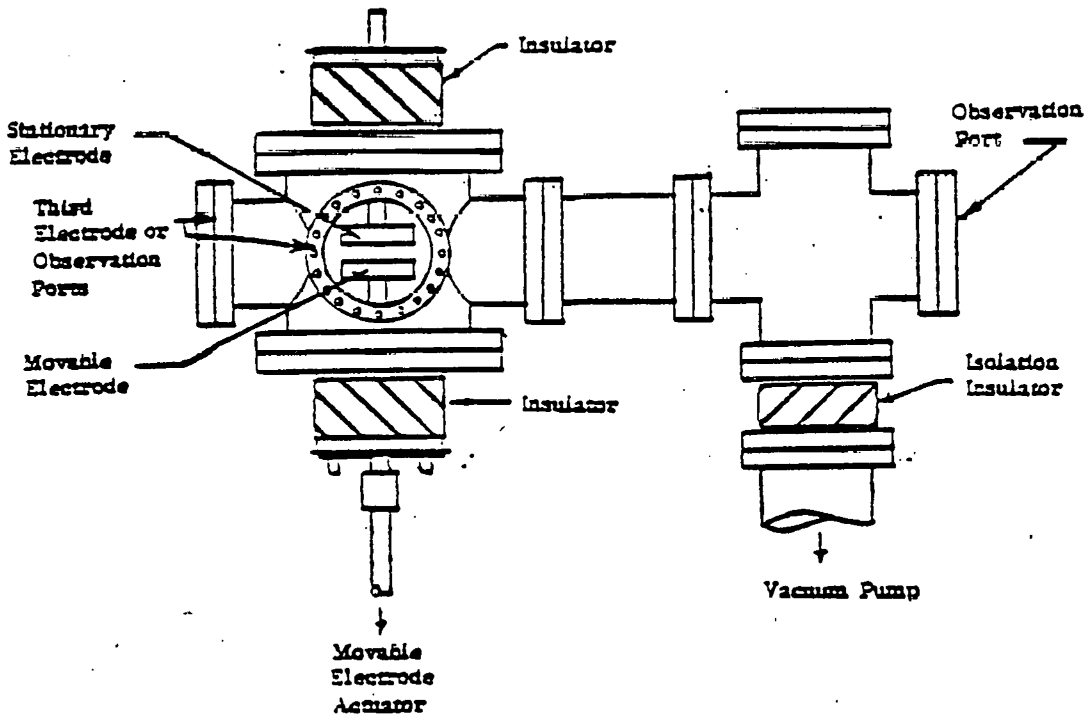


Figure H-20. High Voltage Demountable Vacuum Arc Study Chamber

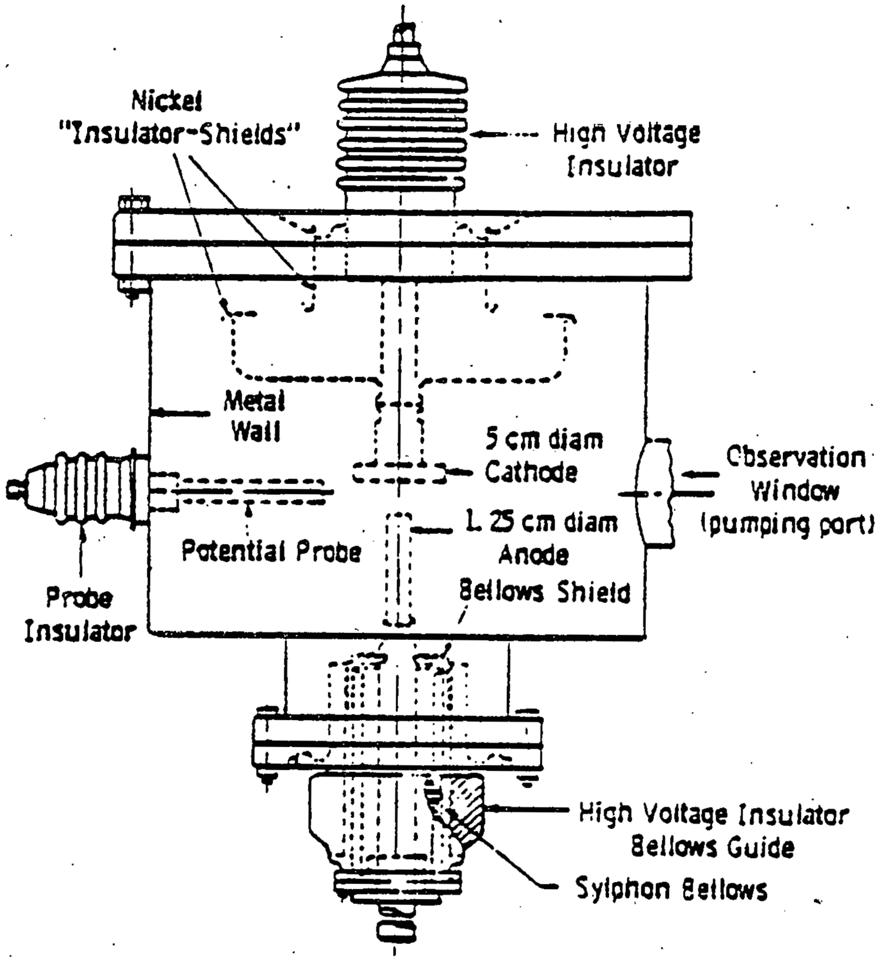


Figure H-21. Experimental Metal-Walled Arc Chamber (Wall Radius 13.5 cm).

via high-voltage insulators. The chamber is equipped with a 5 cm diameter observation window; a bellows permits a maximum electrode separation of 3.5 cm. After bakeout at 300°C, the chamber pressure is maintained at about 5×10^{-8} Torr. In previous studies, this chamber has been used to detect the factors which affect anode spot formation in vacuum; here the anode diameter was varied. Cathode spot properties have also been investigated by measuring the current of metal vapor ions impinging on the chamber walls from the vaporizing cathode.

HIGH VOLTAGE BREAKDOWN FACILITIES

Two experimental chambers, one for voltages up to 100 kV and the other for voltages up to 250 kV, have been designed at Westinghouse and are presently in operation. Both chambers are bakeable to 450°C and each is coupled to an ultrahigh vacuum system capable of evacuating to a residual pressure 10^{-9} Torr. Both chambers are equipped with a bellows device for varying the electrode separation, and with electrode heaters for both outgassing the electrodes and controlling their initial temperature during experiments. In addition, the chambers are equipped with four mutually perpendicular viewing ports with sapphire windows to allow processes occurring between the electrodes to be spectroscopically measured. Alternatively, the ports may be used to couple other equipment to the experimental chamber such as, for example, electromagnetic pole pieces to allow measurements in the presence of magnetic fields.

Two regulated DC supplies are available: (a) 90 kV, 1 mA, 1% regulation; and (b) 250 kV, 10 mA, 0.01% regulation. Two step-function impulse generators are available: (a) A vacuum tube generator of maximum amplitude 20 kV, 0.2 µs rise time, pulse duration up to 1 ms: and (b) a coaxial cable generator of 100 kV maximum amplitude, 5 ns rise time, pulse duration up to 1 µs.

Diagnostic capabilities include facilities for pre-breakdown current measurements down to 10^{-13} A and standard dual-beam oscilloscopic studies.

H.11 LIQUID METAL TECHNOLOGY AND FACILITIES

The near-term thrust of the fusion program is toward water and/or helium as the reference coolants for first wall blanket and shield components and solid. lithium bearing materials for tritium breeding media. Liquid lithium still remains, however, a candidate cooling and breeding medium for ultimate power reactor applications and it would be prudent to consider the possibility that engineering testing of components cooled by liquid lithium will be of long range interest under Test Program Elements I, II and III. A discussion of Westinghouse capabilities in this area is included to indicate our interest in supporting the program in this area.

As a leading participant in the LMFBR program, Westinghouse has one of the largest arrays of in-house liquid metal research facilities in the nation. During the past four years Westinghouse has been active in applying this wealth of experience to the problems associated with the use of liquid lithium in both the magnetic and inertial confinement fusion programs. A number of specialized facilities have been set up at Westinghouse expense to promote research in this area. These facilities are immediately available to serve the needs of the FW/B/S engineering test program.

A concise list of existing liquid metal facilities at Westinghouse has been included in Table H-1. Table H-2 summarizes some of the key Westinghouse activities and accomplishments in the liquid lithium technology area. An important and growing effort is experimental support for the FMIT program conducted by the Westinghouse Hanford Company (Hanford Engineering Development Laboratory).

Experimental work in lithium is centered at two locations: the Lithium Facility for fusion-related Experiments (LIFE) located at the Advanced Reactors Division in Madison, Pennsylvania, and the liquid metals laboratory at the Westinghouse Research and Development Center.

TABLE H-2

EXPERIMENTAL LITHIUM EXPERIENCE AT WESTINGHOUSE

- LITHIUM VAPOR TRANSPORT PREDICTED AND MEASURED FOR SCALED FMIT ACCELERATOR DRIFT TUBE
- LITHIUM IONIZATION DETECTOR DEVELOPED AS A SAFETY ANNUNCIATOR FOR FMIT
- LITHIUM CONCRETE REACTIONS STUDIED BY REACTION CALORIMETRY
- LITHIUM VAPOR/AEROSOL - LASER INTERACTIONS
- LITHIUM COMPATIBILITY WITH POLISHED SIC LASER MIRRORS
- MODIFIED WESTINGHOUSE COMMERCIAL ON-LINE OXYGEN AND HYDROGEN METERS SHOWN FEASIBLE FOR LITHIUM APPLICATION
- ON-LINE NITROGEN METER CONCEPTUALIZED AND SCHEDULED FOR TEST
- LITHIUM FALL EVALUATED FOR TOKAMAK POLOIDAL DIVERTOR PARTICLE TRAP
- EXPERIMENTAL MEASUREMENT OF RADIANT ENERGY DEPOSITION IN
  FLOWING LITHIUM

R&D CENTER LIQUID LITHIUM FACILITIES (LLP)

The liquid lithium facilities at the Westinghouse Research and Development Center contain. among other equipment, a stainless steel. all welded. electro magnetically pumped recirculation loop connected to an adjacent glovebox. The loop, shown in Figure H-22. contains approximately 45 liters of lithium and is designed to operate at temperatures up to 500°C. Impurities are controlled via a flow-through filter. cold trap and titanium getter trap and can be monitored by continuous on-line meters and by intermittent chemical analysis. The development and testing of these impurity monitoring and control techniques are a major feature of planned experimental programs.

The lithium loop has been designed to provide a versatile facility capable of performing many different types of studies in support of fusion reactor blanket development. Part or all of the main flow can be diverted to pass through the glovebox and then back into the loop. In this way, unusual test requirements can easily be accommodated without undue modification of the loop pipework. One example of this is the current test program being performed in support of the FMIT project. In this experiment. lithium flow passes through a rotating disk apparatus capable of simulating the very high liquid velocities expected in the FMIT and thereby enabling measurement of the previously unknown erosive properties of lithium. Other tests, e.g., the study of lithium jets or falls. on-line processing, or interaction with magnetic field lines. could easily replace this erosion test assembly.

Another test stand in the liquid lithium facility includes an operating 2 x 10^{-6} Torr vacuum glove box (1.2 m dia x 2 m long) with numerous ports and flanges. The box is water cooled through a double jacketed chamber. A forced flow lithium loop will be installed through the vacuum glove box early in 1981.

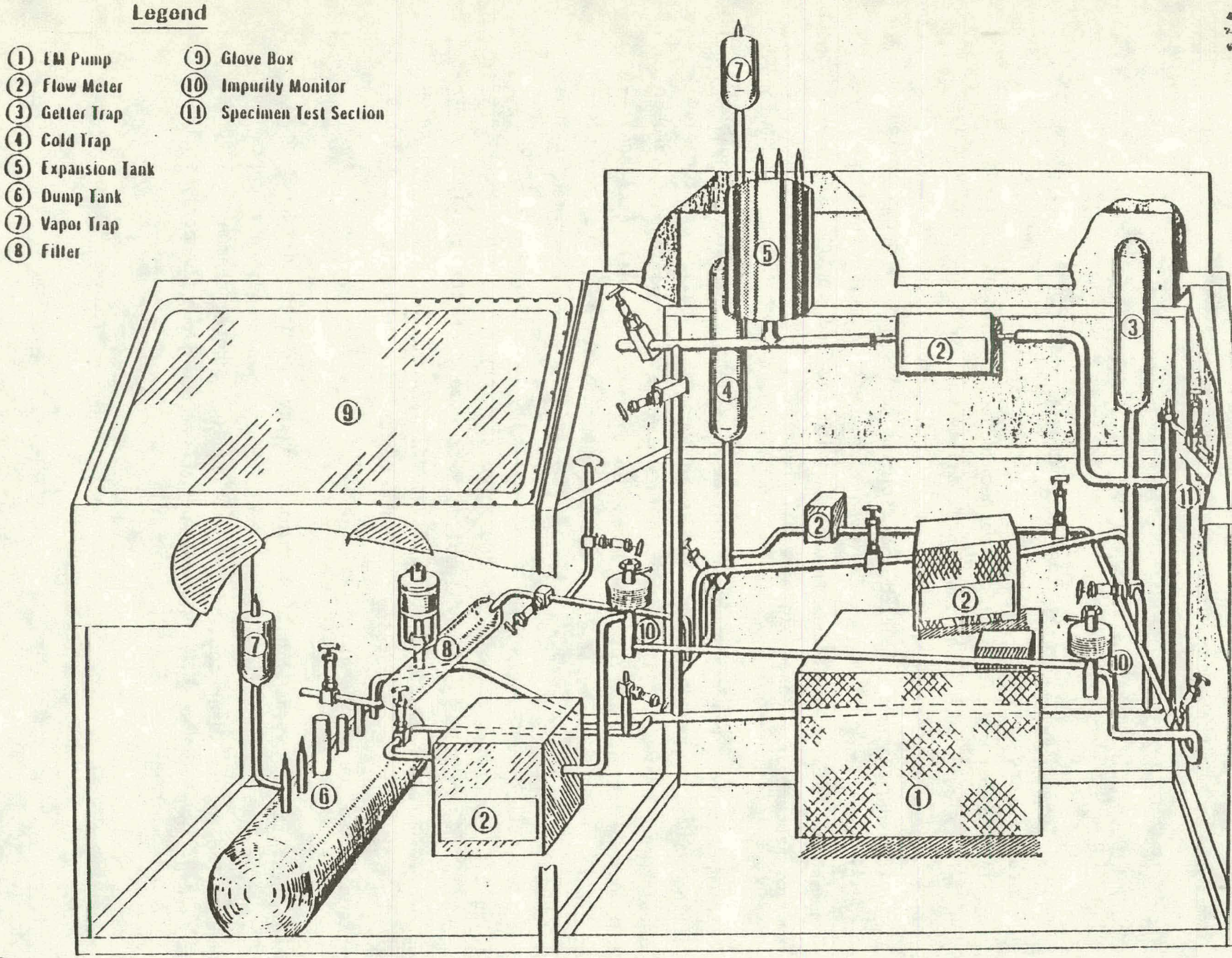


Figure II-22. Cutaway View of the Liquid Lithium Loop Presently Operating at the Westinghouse Research and Development Center

ADVANCED REACTORS DIVISION LIQUID LITHIUM FACILITY (ELF)

In anticipation of a growing interest in first-wall feasibility studies for inertial confinement fusion reactor design, a test program was initiated at WARD (Waltz Mill Site) in 1978 to build and operate a lithium test facility. This initial investment resulted in government funding to perform aerosol tests in collaboration with the Westinghouse Research and Development Center, in addition to the basic corrosion studies. The aerosol work was successfully completed in 1979. The corrosion program which is equally applicable to magnetic confinement concepts, was concluded in June 1980 with additional information obtained from a destructive examination of the test loop (ELF) containment piping, the hot trap and the cold trap.

As a result of information gathered from the lithium community over the past several months, plans for a redesigned facility have been finalized. During this period, it became apparent that the present test loop ELF was not amenable to the modifications required and that in fact, it was more valuable as a source of material for future study and evaluation [T304 SS with up to 2000h exposure to flowing lithium at temperatures in the range 200°C-500°C (392°F - 1022°F].

The following action to date has, therefore, been taken under a program funded entirely by the Corporation:

- The ELF loop has been dismantled and placed in storage. Two small pumps and the main loop heat exchanger will, however, be reused;

- A new facility has been designed: the Lithium Facility for Fusion-related Experiments to be known by the acronym "LIFE".

The LIFE loop is now about 70% complete and, after shakedown tests, will be ready to commence corrosion studies by June 1981. This new work will be directed particularly towards the following:

- Continued corrosion evaluation at 550°C. Exposure of samples in a low temperature test leg (<300°C) will be initiated later, also;
- Weldments, valve facing materials, etc:
- Austenitics vs Ferritics;
- Meter studies for hydrogen and nitrogen;
- Liquid metal analysis techniques;
- Hot trap, cold trap, magnetic trap studies;
- Velocity effects on corrosion rate;
- Stress effects on corrosion rate.

The main loop is constructed of 1.00 inch T309 tubing and contains an E-M pump (MSAR Style VI: 2 gpm capacity), a flow meter, a tube-and-shell exchanger, a test vessel to which the isothermal and the deposition (decreasing temperature) test legs are attached, a magnetic trap and a finned cooler. The test vessel will also act as the expansion tank and will be used for regular lithium sampling for chemical analysis.

Side loops contain separately pumped hot trap and metering circuits as well as a bypass flow cold leg containing a cold trap. It is planned to use this cold leg later in the program for low temperature corrosion studies.

The objectives of the follow-on program are threefold:

- to extend the corrosion data base beyond 2000h at 1000°F/550°C for a variety of candidate first-wall and magnetic fusion containment materials;

- to gain additional operating experience with a lithium test facility;

- to develop methods of controlling, metering, and measuring the level of hydrogen and nitrogen, in particular, in liquid lithium.

H.12 THE HIGH ENERGY ION BOMBARDMENT SIMULATION FACILITY

The High Energy Ion Bombardment Simulation Facility (HEIBS) is a unique example of university-industry cooperation in applied research. The facility is located at the Sara Mellon Scaife Nuclear Physics Laboratory of the University of Pittsburgh and is operated jointly by the University of Pittsburgh and Westinghouse.

Work was initiated on the facility in 1979 with private funding. Subsequent operation has been supported internally (faculty-student research and Westinghouse research) and a variety of government grants (DOE, NSF). The primary thrust of research at the facility is the simulation of materials damage due to high energy neutron irradiation through ion implantation.

The basic technique employed is coimplantation; heavy ions provide atomic displacement effects and implanted helium simulates accumulation of He through (n,α) reactions. A key aspect of the research is the study of swelling and irradiation induced microstructural changes in nickel based alloys for fusion applications as part of the OFE-ADIP program.

The overall laboratory set-up for co-implantation experiments is schematically depicted in Figure H-23 which shows the four floors of the laboratory. On the first floor heavy negative ions are generated in a special cesium sputter-ion source. Details of the "UNIS" source and the optics which provide optimum matching to the E-22 tandem accelerator are given in Figure H-24. For many of our experiments a Si beam is used to produce displacement damage in the sample and we have therefore chosen Si as our example in Figure H-23. The Si^-beam is mass analyzed by magnet M_2 prior to injection into E22 with a kinetic energy of 70 to 100 kV. Since we must reach a compromise between depth of penetration and heating of the target, we have found that a final beam energy of 28 MeV is very satisfactory. This gives a mean depth of penetration in 304 SS of 4.67 μm while the peak of the displacement damage occurs at 4.5 μm from the surface. To attain this energy the E22 terminal is set at a

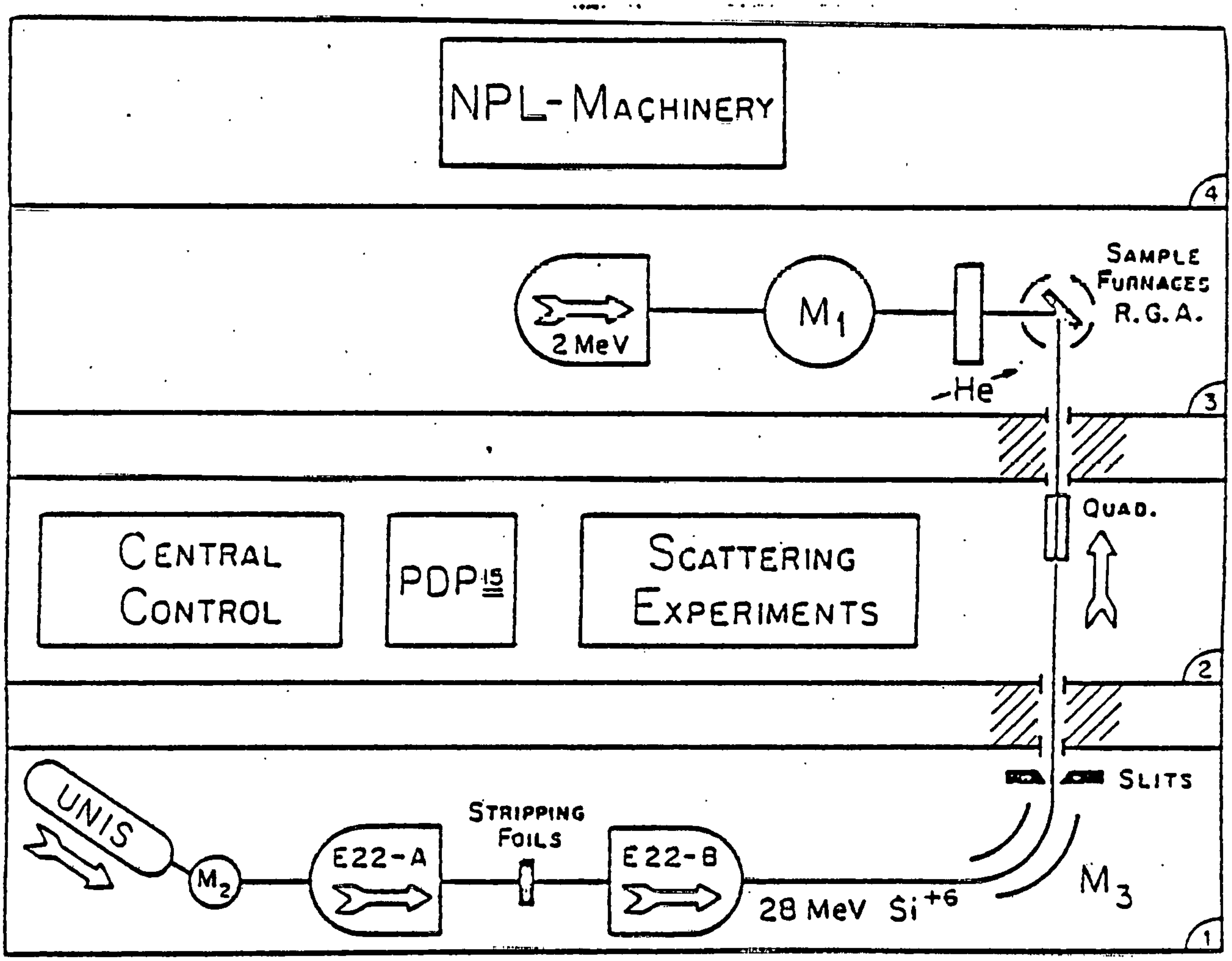


Figure H-23. Overall Elevation View of the HEIBS Facility Showing Major Beam-Line Components. Heavy ions are generated by a negative ion source (UNIS) and accelerated by the E-22 Tandem Accelerator. The light dopant ions are generated in a 2 MeV Van de Graaff generator on the third floor of the building.

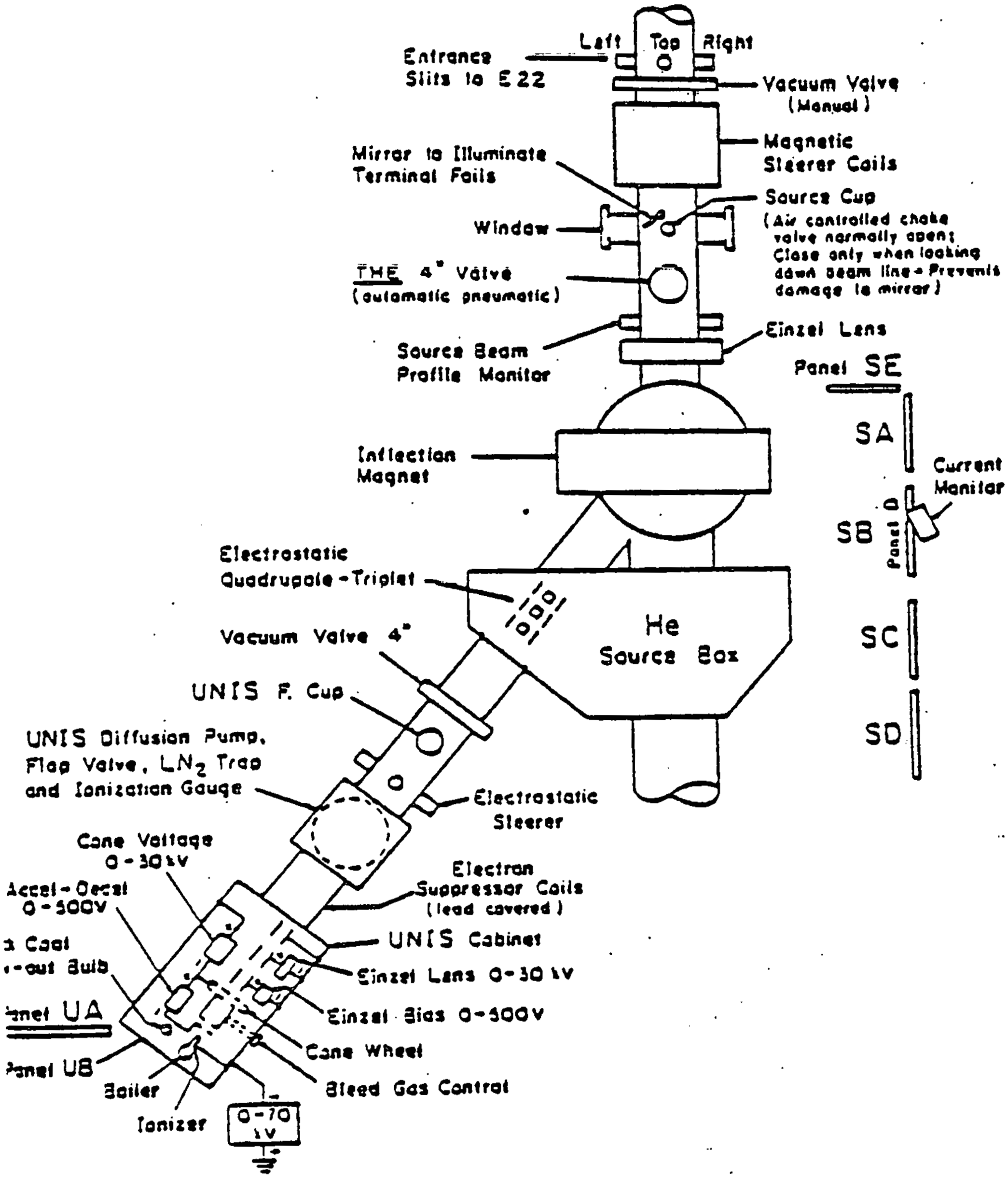


Figure H-24. Detailed Schematic of the UNIS Ion Source and Beam-line Optics Leading to the Tandem Accelerator E-22.

positive potential of 4 MV. At the terminal the 4 MeV Si^- ions pass through special carbon stripping foils. These 250 Å thick foils are made by our group at the Westinghouse R&D Center using AC sputtering of ethylene. Compared to evaporated carbon foils which are commercially available, these foils have an order of magnitude better life-time under heavy ion bombardment. At 4 MV the major positive charge state which is accelerated in section E22 B of the tandem is Si^{+6}. Consequently, an emerging Si^{+6} ion will have an energy of 28 MeV. Transmission through the accelerator is nearly 100%. Since many positive charge states are created, a large 90° analyzing magnet (M_3) in conjunction with slits is used to ensure that the beam consists exclusively of Si^{+6}. The analyzed beam now travels to the second floor where it is refocused and passes to the third floor target chamber. Beam diagnostics are performed all the way from UNIS on the first floor to the target on the third floor. The target itself is located inside two furnaces which are feedback controlled to ensure that beam fluctuations do not cause major fluctuations in temperature. The 3 mm samples undergo an elaborate mounting procedure to ensure that they remain within 1°C of the temperature which is actually measured. In general we can maintain a temperature in the range of 550 C to 750°C - 2°C. An additional feature of the all metal target chamber is a high resolution residual gas analyzer made by the Extranuclear Corporation. This enables us to monitor the ambient vacuum conditions around the sample. Vacuums of 10^{-7} Torr are achieved during implantation by means of two 450 liter per second turbo pumps located on either side of the target assembly.

The dopant ions are generated in a 2 MV Van de Graaff generator on the third floor of the building. Beams are mass separated by a large magnet. Steering, rastering and diagnostics are available along with a "doping" beam line. Prior to entering the sample chamber the beam can be "shaped" in energy in such a way as to dope the target over a considerable depth. In the case of He, for example, we can dope in a predetermined way from 0.5 µm to 3.5 µm below the surface. All experimental parameters during an experiment are recorded.

At times it is desirable to use nuclear techniques for profiling dopants in targets. If the energies required are such as to pose a radiation hazard, such experiments are done in the shielded second floor area. A PDP-15 computer and PACE interface is available for on-line processing of the scattering data.

On the third floor there are three additional experimental setups which are indicated in schematic form in Figure H-25. The Rutherford backscattering or channeling experiments use a He+ beam from the Van de Graaff with an energy ranging from 0.5 MeV to 2.0 MeV. A 4.8 meter channeling line gives a full width angular divergence of $\sim 0.03°$. Our goniometer has two angular degrees of freedom, both with stated accuracies of 0.01°, and one translational degree of freedom. Recently, the whole system has been interfaced with an LSI-11 computer. Plots of the ratio of channeled to random yield as a function of depth for a particular matrix and He energy are available in minutes after the accumulation of adequate points for good statistics. Automatic searching with the aid of the computer for an absolute minimum in the angular yield profiles, for a given crystal direction, has been achieved. Automating this procedure improves the accuracy and saves a considerable amount of time.

Static and dynamic mechanical experimental apparatus are available; a uniaxial tension head or a moving sample head go into the beam line marked "Mechanical Relaxation" on Figure H-25. The "heads" have also been made compatible with the tandem accelerator beam line so that either <2 MV protons, helium ions or high energy heavy ions can be used to implant the specimen.

Finally, the position marked "UNIS-test" of Figure H-25 represents a fully instrumented negative-ion source test bench. Such a test bench is extremely valuable for determining the parameters required for generating a negative ion beam.

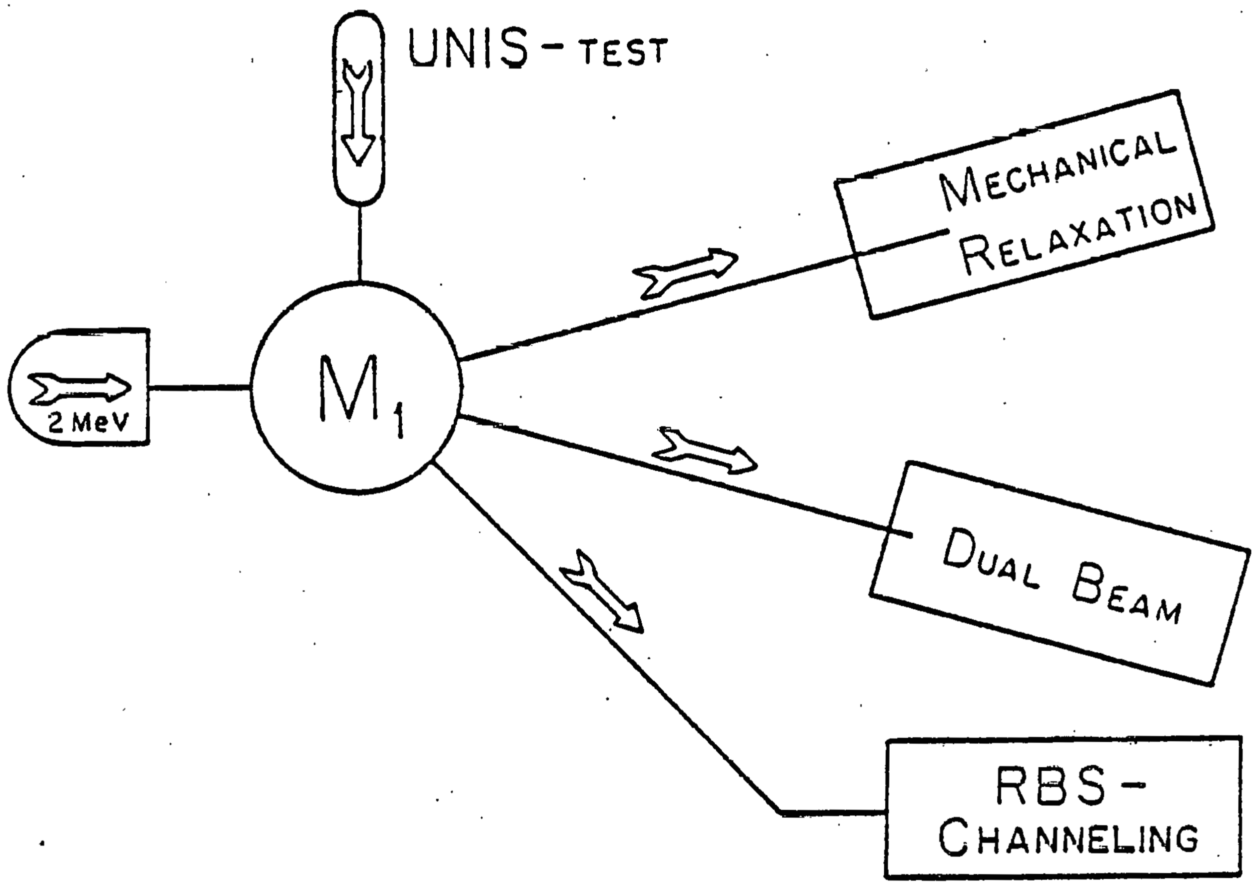


Figure H-25. HEIBS Experimental Beam-lines for Rutherford Backscattering or Channeling Experiments, Dual Beam Experiments, Static and Dynamic Mechanical Tests and Negative Ion Source Experiments.

H.13 Computers and Computational Tools

Our team has at its disposal a complete range of digital and analog computational facilities (and codes as tabulated in Appendix A) which are fully rele- vant to the needs of the First Wall/ Blanket/Shield Program. The facilities available include two large scale centralized systems: the Westinghouse Nu- clear Energy Systems Computer Facility in Monroeville, Pennsylvania, and the National Magnetic Fusion Energy Computer Center (NMFECC) at Livermore, Cali- fornia. Both are accessible by remote terminals as well as local batch facil- ities serving the activities supporting the project team at various sites. These computer facilities support programming activities with the most popular languages including FORTRAN COBOL APL ALGOL and/or BASIC besides some others less frequently used. Capability for both batch and interactive modes of operation, besides extensive software libraries and graphics options, are also present at these facilities.

WESTINGHOUSE NUCLEAR ENERGY SYSTEMS (NES) COMPUTER FACILITY

The NES Computer Facility was established at the Monroeville Pennsylvania, Nuclear Center to provide a powerful centralized computational capability to serve both the commercial nuclear power divisions and the advanced power sys- tems divisions. These facilities are linked to 19 separate sites via high speed remote terminals. The overseas operations in Spain and Belgium are linked via satellite. At NES, a CYBER 173 supervises the workload of two CDC 7600's that handle the scientific calculations for most of the Corporation. An IBM 3033 responds to the business needs and some special scientific appli- cations. An INTERCOM system on the CYBER 173 can be employed in the interac- tive mode for file manipulation in the preparation of input for batch process- ing on the CYBER 173 or on the CDC 7600s. INTERCOM can be used with any re- mote teletype-compatible ASCII terminal, including CRT display, printing and graphics terminals. Also at NES, the printed and graphics output may be ob- tained in microfiche form as an alternative to hard copy for long term stor- age. As in any large scale computer facility there is an extensive system of support I/O devices. A competent and responsive staff complements the hardware capability and is prepared to advise when difficult programming problems occur.

NATIONAL MAGNETIC FUSION ENERGY COMPUTER CENTER

Westinghouse has an account with NMFECC at Livermore. California, and. consequently, access to the code library and extensive computing capability of the CDC 7600 and CRAY machines at this nationally recognized computing network.

RESEARCH AND DEVELOPMENT LABORATORY COMPUTER CENTER

The Westinghouse R&D Laboratory maintains for its users a UNIVAC 1106 and ancillary I/O support equipment including graphics with a CALCOMP 936 plotter. The Conversational Time Sharing system installed on the UNIVAC is interactive and permits remote job submittals.

Also located at this site are two separate interactive graphics systems for computer assisted drafting and design to provide rapid and easy design capability. The Applicon System is employed most frequently for electronics applications while the Computer Vision interactive graphics system receives the bulk of the mechanical design applications. A full complement of hardware. software and skilled engineers and technicians support the use of these facilities.

AESD (LARGE SITE) HYBRID COMPUTER FACILITY

The hybrid computer laboratory at the Advanced Energy Systems Division (AESD) Site in Large, Pennsylvania is used primarily for analyzing the time dependent behavior of emerging energy system concepts. In addition, the digital computer section can be used separately as a scientific digital computer to perform calculations or to process data.

The analog computer consists of three EAI 2000 analog computer consoles. Each console can solve up to 31 simultaneous differential equations and all three can be combined to triple that capacity. Experience has shown that this is sufficient capacity to handle most problems in the energy field. The power of the analog computer is multiplied by the hybrid interface to the digital computer. The digital computer can be used to calculate complex functions such as water properties for use by the analog computer. The hybrid interface has 32 analog-to-digital channels and 16 digital-to-analog channels.

Use of the hybrid computer is facilitated by the recent developments in computer software which allow automatic setup, scaling and checkout of the analog computers from the digital computer. This reduces both the engineering time and the calendar time required to develop a simulation. Evaluation of results is aided by a digital graphics system which can instantly display results of a transient on a CRT, scaled and labeled in engineering units. A hard copy of the graphics display can be obtained simultaneously and its quality permits direct use in report preparation.

The digital computer is an SEL 32/77 minicomputer with a 32 bit word and a 64 K word memory. The computer is capable of timesharing operation using the three terminals in the system, batch operation from cards, or hybrid operation with the analog computer. Mass storage is provided by a magnetic tape drive and a 40 M byte disk. Output is via a 600 LPM printer. The primary programming language for this computer is an extended version of FORTRAN IV, and a large scientific subroutine package is included.

This computational facility is available to the First-Wall/ Blanket/Shield program as required, particularly for the analysis of complex transient behavior of components and subsystems.

H.14 HEDL FACILITIES

Table H-3 is a summary of existing test facilities at the Hanford Engineering Development Laboratory which could be used to support the FW/B/S program. While the use of these facilities is not contemplated at this time, some of the capabilities may be suitable in special applications should a significantly expanded TPE I program, requiring accelerated and multiple parallel testing capabilities, materialize.

TABLE H-3

EXISTING HANFORD ENGINEERING LABORATORY (HEDL) TEST FACILITIES

FACILITY	COOLANT	FLOW	PRESSURE	TEMPERATURE	PURITY CONTROL	POWER SUPPLY	HEX CAPACITY	OWNER	CUSTODIAN
Thermo-mechanical (1)	Water	0-15000 gpm	0-250	70-250°F	PH	--	Feed & Bleed 300 gpm	DOE	HEDL
High Pressure	Water	0-250 gpm	0-2500 psig	70-600°F	PH, O_2	4 MW 32 K amp	4 MW	DOE	PNL
Transient Test Loop	Sodium	0-600 gpm	0-300 psig	300-1200°F	Cold Trap <1 ppm O_2	1.4 MW 18K amp	5 MW	DOE	HEDL
Experimental Lithium System (2)	Lithium	0-600 gpm	0-50 psig	400-800°F (4)	Cold Trap Hot Trap <100 ppm N_2	--	(3)	DOE	HEDL
Inert Gas Loop	Helium	0-4 lb/sec	0-465 psia	420-660°F (5)	None	--	0.5 MW	DOE	HEDL

GENERAL NOTES:

1. A portable electron beam power supply system is common to all facilities.
2. Under power supply heading, these are controllable DC supplies in addition to the power supply in Note 1. These supplies are capable of transient power cycling.
3. Future expected use of these facilities is low enough that no scheduling conflicts are expected which would prevent running substantial FW/B/S tests in the time period proposed.
4. Addition of a shell and tube hex at low cost would extend heat dissipation to any desired level. Addition of a heat exchanger is required to match heat input. Existing heat removal may have to be increased.
5. Can easily upgrade to at least 1000°F. Limit applies to He compressor. Higher test section outlet temperatures can be obtained by adding a heat exchanger downstream of test section.

Fusion Power Systems Department • Fusion Power Systems Department • Fusion Power Systems Department • Fusion Power Systems Department

APPENDIX F

APPENDIX F

COMBUSTION ENGINEERING INC. RESPONSE

C-E Power Systems
Combustion Engineering, Inc.
1000 Prospect Hill Road
Windsor, Connecticut 06095

Tel. 203/688-1911
Telex: 99297

POWER SYSTEMS

September 24, 1981
EDS-81-231

P. Y. Hsu
TPE-II Program Manager
Fusion Technology Program
E. G. & G. Idaho, Inc.
P. O. Box 1625
Idaho Falls, Idaho 83415

Subject: Thermal Hydraulic Thermomechanical Testing Facilities Survey

Dear Mr. Hsu:

Combustion Engineering, Inc., is pleased to respond to your inquiry with regards to available C-E facilities to perform thermal-hydraulic and thermo-mechanical testing planned for the fusion technology program.

The attached response addresses each of the specific items in the inquiry list contained in your letter to V. G. Scotti, dated September 11, 1981, that we feel we are qualified in providing. C-E has the requisite expertise and facilities to make an important contribution to the fusion program.

We would be pleased to provide further amplification of our capabilities. Please call Dr. H. N. Guerrero at (203) 688-1911 ext. 3921 for any questions.

Very truly yours,

Bernd J. Selig
Director
Engineering Development & Services
C-E Nuclear Power Systems
Combustion Engineering

COMPONENT TESTING

The technical responsibility to test major C-E NSSS components in order to establish design parameters or to verify the completed design is assigned to the Engineering Development and Test group (ED&T). The ED&T group is staffed by 23 engineers and specialists and is one of the functional groups under Engineering Development and Services.

To meet the above responsibility, scaled and full size models of C-E's PWR primary components (fuel assemblies, CEDMs and primary system internal structures) are tested to support their thermal, hydraulic and mechanical designs. Further, prototype components such as fuel assemblies and CEDMs are tested at prototypical conditions of pressure, temperature and flow to assure that stringent performance criteria are met.

Besides operational testing, the ED&T group also performs research and development testing to: resolve nuclear safety concerns such as LOCA heat transfer and blowdown loads in the core and steam generator; conduct flow and vibration studies of major internal structures; conduct testing for corrosion effects; and develop specialized single and two-phase flow instrumentation.

TF-7 Control Element Drive Mechanism (CEDM) Test Stand

This facility is used as a control element drive mechanism test facility for hot operational development testing of CEDMs, control elements, and CEDM cooling shrouds. The facility has a 20 foot and a 31 foot head clearance area.

Pressure	-	2500 psi
Temperature	-	650°F
Flow	-	30 gpm
Material	-	Stainless Steel
Test Sections	-	8 inch diameter
	-	16 inch diameter
Piping Size	-	1-inch

TF-2 Hot Loop

The CE reactor components test facility (TF-2) is a high temperature/high pressure test loop that is similar in many respects to a PWR coolant loop. It's primary use is for simultaneously testing full size systems of reactor components including up to five fuel assemblies, control rod, control rod drive mechanism and reactor internals mock-up. In addition, high temperature/high pressure model testing and individual component testing is carried out in this facility. The facility has been employed to develop an ultrasonic flow meter, perform special erosion tests, conduct leak detection studies and serve as a materials evaluation station.

Pressure	- 2500 psi
Temperature	- 650°F
Flow	- 15,000 gpm at 300 ft. head
Piping Size	- 16 inch
Test sections	- a. 36 inch diameter, approx. 40 ft. long b. 13 inch diameter, approx. 28 ft. long
Material	- Stainless steel 316 and stainless clad carbon steel for the 36" test section

TF-1 Hot Loop

TF-1 is a 4" high pressure, high temperature flow loop used for (1) vibration and dynamic corrosion testing of partial fuel bundles, (2) LOCA blowdown loads testing of full length steam generator tubes under prototypical conditions, (3) blowdown heat transfer testing of simulated PWR fuel channel under simulated transient flow and power conditions.

Pressure	- 2300 psi
Temperature	- 650°F
Material	- Stainless Steel
Flow	- 450 gpm at 150 ft. head
Auxiliary Power Supply	- 300 kw DC or AC programmable

TF-16 Pot Boiler Facility

The pot boiler test facility offers the capability of testing a variety of chemistry/corrosion environments under heat transfer conditions. Pot boilers are designed to simulate extreme steam generator hydraulic conditions. Testing in this facility included evaluation of various steam generator secondary chemistry environments under normal or faulted conditions and simulated condenser leakage tests. Other efforts include denting neutralization, chemical cleaning, condensate polishing, water chemistry limits, and alternate materials.

Pressure	- 2500 psi primary (1000 psi secondary)
Temperature	- 650°F primary (550°F secondary)
Flow	- 150 gpm at 310 ft. head
Material	- Stainless steel primary, (carbon steel secondary)
Test Station	- 17 (one sta. has 250 kw rating)

SPECIAL INSTRUMENTATION

Examples of special instruments developed for the Nuclear Laboratory test facilities and reactor use are:

- Boronometer to measure B^{10} concentration in the reactor coolant.
- Ultrasonic flowmeter to measure primary coolant flowrate.
- Ultrasonic displacement measuring system for core barrel vibration measurements.
- RF reflectometer probe for dynamic liquid level measurement.
- Control rod position indicator system.
- Precritical Vibration Monitoring Program system consisting of accelerometers, pressure transducers and strain gauges.

ELECTRICAL AND INSTRUMENTATION DEVELOPMENT

The Power Systems Group of Combustion Engineering is involved in a wide range of activities which require electrical and instrumentation support. These needs are met by the Electrical and Instrumentation Development group within the Engineering Development and Services organization. Staffed by eight engineers, nine electronic technicians and supervisor, personnel, the group also maintains a fully equipped electronics lab for developing prototype instruments and a standards lab for calibration of a wide variety of instrumentation.

DATA ACQUISITION AND PROCESSING FACILITY

The Nuclear Development Laboratory data acquisition/data reduction systems are centered around a PDP11/15 minicomputer. Data acquisition equipment includes two 14 track FM tape recorders, a 39-channel FM Multiplex system, and a direct digital data acquisition system tying the PDP computer to test stands. The PDP-11 computer system contains 96 channels of analog-to-digital conversion, ten million bytes of on-line disc storage, seven-track digital tape, a 132-column line printer and a digital incremental plotter for automatically generating report quality plotted data. Remote and local access exists to C-E's data processing center which includes CDC 7600 and Cyber 72 computers. Standard Fourier analysis and statistical capabilities are available.

C-E RESPONSE TO QUESTIONNAIRE FOR TPE-II FACILITY SURVEY

Reference: Letter from P. Y. Hsu, TPE-II Program Manager, Fusion Technology Program to V. G. Scotti, September 11, 1981.

C-E's response is directed mainly to the thermal hydraulic testing aspect of TPE-II. Here, we assume the testing of simulated internal heat generation in facsimiles of fusion blanket/shield modules while connected to and actively cooled by a heat transfer loop. Of course, this may also be considered as constituting part of the thermo-mechanical testing. The following items address each of the specific questions in the inquiry of the above reference. (The numbering or letters correspond to the sequence in the reference letter.)

Question 2

a.) The Nuclear Laboratories presently has a 300 kw fully programmable DC power supply (107V DC, 3000 amps), where the output is continuously variable from 10 volts to the maximum voltage. This has been used previously to power simulated nuclear fuel rod assemblies for critical heat flux, blowdown and transient heating experiments. In addition, the Laboratory has available up to 1.87 MW additional electrical capacity with the requisite transformers and switchgear. Thus, high power loadings of up to 2.17 MW can be accommodated for steady operation. Cyclic or transient experiments can be performed at up to 300KW with the DC programmable power supply which could be augmented by addition of similar units.

Resistive heating utilizing a low voltage, high current power supply would appear to be the most technically feasible method of simulating the internal heat generation of a blanket module since a lot of experimental experience has been achieved in simulating nuclear heating in fission reactors with this method. The technology is well developed and various power profiles can be easily generated by tailoring the resistance per unit length of the resistance heater. This is a highly flexible method which can accommodate any conceivable combinations or configurations of blanket modules.

b.) The test environment in which the test module would be set up would normally be an air environment.

c.) The size and configuration of the test pieces would be limited only by the heat source power and the test piece heat generation/unit volume.

d.) Any material can be accommodated in the test space if safe working conditions can be maintained.

e.) Maximum temperatures would be limited by the meeting point of structural materials (e.g. steel ~ 2000^{o}F). Maximum power attainable as mentioned previously is 2.17 MW.

f.) The rise time and decay time of the DC power supply is in the order of 1/4 second. However, the control current of the saturable reactor core is only 0-100 ma which easily allows the programming of any power pulse rise and decay time and repetition rate using simple electronic control circuits.

g.) The C-E Nuclear Test Laboratories include a number of general purpose high pressure, high temperature water loops that operate at PWR reactor conditions of 2200 psig and 600^{o}F. Three of these that may be pertinent to these tests are:

a. TF-1 loop with a flow capacity of 400 gpm and 4-inch pipe line size;

b. TF-2 loop with a flow capacity of 15,000 gpm and 16-inch pipe line size and,

c. TF-16 loop with a flow capacity of 150 gpm and 3-inch pipe line size.

Detailed specifications and description of these test loops are included in Attachment A. Flowsheets are included in Attachment B.

h.) All of these facilities are fully instrumented for loop pressure, temperature and flow operational conditions. Special instrumentation of test sections installed in these loops have been both varied and substantial. These include fast response thermocouples, pitot probes, ultrasonic flowmeters, piezo-electric pressure transducers, piezo-electric and strain gage type force transducers, eddy current displacement transducers, LVDTs, strain gages, accelerometers and others. The special circumstances under which these types of instrumentation and the expertise developed with them can best be illustrated with examples of testing conducted with these facilities. These include:

- Hot flow testing under simulated reactor primary system environments, testing under simulated seismic conditions, equipment evaluation of typical full size reactor components including reactor fuel, control elements, drive mechanisms and miscellaneous support structures.

- Flow testing on 1/5 scale models of complete reactor vessel and all its internals for verification of thermal-hydraulic analyses.

- Flow induced vibration tests of a scale model PWR Upper Guide Structure under prototypical reactor pressure, temperatures and flows.

- EPRI Blowdown Heat Transfer tests and other reflood tests of simulated single reactor fuel channels.

- EPRI LOCA loads testing of full size PWR steam generator tubes under prototypical reactor blowdown conditions for verification of thermal-hydraulic code (CE-FLASH4) and mechanical response codes (ANSYS).

- EPRI Steam Generator Tube fretting and wear test program to characterize tube/support interaction forces as a function of tube/support geometry, loading conditions, environment and materials.

A central data acquisition system based on a PDP 11/15 computer is available for on-line remote data acquisition (up to 50,000 samples per second) as well as off-line data reduction. An on-line data acquisition capability would have advantages for the TPE II program in view of the capability for data logging as well as transient data acquisition (2 speed flexibility), and capabilities for pre-test calibration, check-out of instruments, and quick turnaround of data reduction. A larger, more flexible computer system is also planned which can handle multiple tasks. A 50-channel FM-Multiplex data acquisition system is also available.

i.) Availability of these facilities at the present time is excellent. Existing C-E test facilities made available to the FW/B/S program will be made available without any "use charge" or "rental fee". Direct charges will only be applied to the contract for labor, materials and extraordinary fuel expenses. C-E's government approved accounting system is based upon depreciation of capital facilities over their useable lifetime. This depreciation expense, as well as other overhead related expenses, is included in the overhead rate for the cost center serving the particular test facility.

j.) C-E, as a nuclear manufacturer, maintains a number of manufacturing operations, and machine shops which the Nuclear Test Laboratories regularly make use of in the fabrication of test models, other test hardware, prototype equipment and special handling tools and fixtures.

A dynamic test laboratory is maintained which is equipped with hydraulic and electromagnetic shakers controlled by a digital vibration control system. Software is also available for dynamic test analysis. This laboratory would be useful for obtaining the dynamic characteristics of the test pieces prior to testing.

An electronics and standards laboratories are available for calibration and fabrication of special transducers and instrumentation.

A full complement of supporting laboratories including analytical chemistry, metallographic, ceramic, materials properties, x-ray, and photographic laboratories are also available. The above facilities are described fully in Attachment A.

k.) C-E Nuclear Test Laboratories maintain a large staff of well qualified personnel for performing the research and development activities of the company using these facilities. There are up to 62 scientists/ engineers and 70 technicians in the various departments comprising the Nuclear Test Laboratories. The various groups and their functions are described in Attachment A.

1.) In addition to the personnel in the Nuclear Test Laboratories, other qualified personnel are available to provide analytical support to tests of the type in TPE II. Such support can be in the form of help in test facility design, test planning, pre-test analysis and post-test data evaluation. C-E is very well qualified in this respect in view of the expertise available among its various design groups in the C-E Nuclear Power Systems. The engineering, materials and physics problems specific to the magnetic fusion program are similar in many respects to the type of problems that Combustion Engineering is regularly dealing with in designing nuclear generating stations. Therefore, the analytical skills and advanced computational methods and broad technical support from all engineering disciplines that are available within C-E will ensure that the test objectives of TPE II will be successfully addressed. The following are the C-E nuclear design groups and a short description of their functions and capabilities.

The Reactor Design Department is responsible for performing structural analysis of reactor fuel assemblies and reactor internals, considering thermo-structural interactions of reactor components, fluid-structure interaction under static, thermal, steady state and dynamic transient loads resulting from accident conditions. Performance of these analysis requires the knowledge of complex computer codes.

Mechanical Design Department is responsible for performing structural analyses of reactor internals, reactor fuel assemblies and auxiliary reactor components. Specific areas of expertise include, but are not limited to:

(a) Application and development of numerical computer methods for the solution of nonlinear dynamic responses under earthquake and fluid pressure loading. Nonlinear capabilities include treatment of gaps, friction, elastic-plastic properties and fluid structure effects.

(b) Development of large finite element models of complex structures for use in both static and dynamic response analyses.

(c) Transient and steady-state dynamic and stability analyses of arbitrarily loaded shell structures including fluid-structure interactions.

(d) Facility in the application and development of a wide variety of structural analysis computer programs such as SAPIV, STRUDL, STARDYNE, ANSYS, SHOCK, MARC, ASHD, DYNASOR, and others.

The Materials and Chemistry Development Department has responsibilities for selecting all primary, secondary and auxiliary systems structural materials, coatings, and other materials used in the fabrication of reactor internals, pressure vessel, primary system boundary and auxiliary components. It handles programs dealing with radiation effects, radiation surveillance of reactor vessel materials, evaluation of post-irradiated material and properties and fracture toughness testing support of NSSS design development and operation of PWR and other advanced reactor systems. The department is also responsible for NSSS system chemistry, system corrosion and chemical cleaning.

Plant Engineering Department provides the design, analyses, and syntheses of thermal-hydraulic and mechanical systems used by the power generation industry. Under this department, the Structural Systems section is responsible for performing stress, thermal, and seismic analysis of piping, components and equipment supports for both primary and auxiliary system designs. The Primary Systems section provides thermal-hydraulic system responses for steady-state, transient and accident conditions and develops dynamic mass and energy release data for postulated ruptures on high and moderate energy piping systems.

Fuels Development Department is responsible for executing materials development programs and irradiation tests on all core materials for C-E - PWRs. It provides consulting services on clad material properties for other groups within NPS. It prepares design input data and correlations on core materials for fuel performance analysis.

Additional Questions

a.) The three test loops cited are all basically circulation type loops. TF-2 is used for large scale testing of prototype components. TF-1 has been used for blowdown and two phase testing. TF-3 is used to simulate PWR primary and secondary (two-phase) conditions.

b.) Water is the primary fluid used.

c.) TF-2 effects its heat-up by means of pump heat (~ 1 MW) and has a cooldown heat exchanger (liquid-to-liquid) rated at 2.5 million BTU/hr-ft^2.

TF-1 has 75 KW in-line immersion heaters and the 300 KW DC power supply is often used in tests using this facility. Cooling capacity is available with a liquid-to-liquid heat exchanger rated as sufficient to handle a 300 KW heat load.

TF-16 has in-line immersion heaters that add up to a maximum of 500 KW. An air blast heat exchanger is used to condense steam generated in the pot boilers and is rated at 1.5 million BTU/hr.

d.) Detailed information on flow rates, pressures, pressure drops, and other information are contained in specification sheets provided in Attachment A.

ATTACHMENT A

C-E Facilities and Laboratories

APPENDIX G

APPENDIX G

BABCOCK AND WILCOX RESPONSE

Babcock & Wilcox Contract Research Division

a McDermott company

P. O. Box 1260
Lynchburg, Virginia 24505
(804) 384-5111

September 24, 1981

Mr. P. Y. Hsu
TPE-II Program Manager
Fusion Technology Program
EG&G Idaho, Inc.
P. O. Box 1625
Idaho Falls, Idaho 83415

Dear Mr. Hsu:

Subject: Your letter dated September 11, 1981, entitled "Thermal-Hydraulic Thermomechanical Testing Facilities."

Babcock & Wilcox appreciates the opportunity to present our capabilities per your subject request. Included with this letter are the following write-ups and brochures:

1. Existing Thermal-Hydraulic Research Facility for Fusion Blanket/ Shield Test Program (TPE-II)
2. Personnel Resumes
3. Heat Transfer Facility with isometric arrangement and summarized capabilities.
4. Research and Development Brochure

We believe this information provides all the answers to the questions in your letter. However, if we have overlooked some aspect, please contact me, (804) 385-3322, or Mr. Bob Kuchner, (216) 821-9110, Ext. 529), to get any further data.

We look forward to working with you on this important industry program.

Sincerely,

BABCOCK & WILCOX
a McDermott company

Floyd N. Anderson

Floyd N. Anderson
Marketing Specialist
Nuclear Marketing Department

FNA:ps
Enclosures
cc: G. Longhurst

EXISTING THERMAL-HYDRAULIC RESEARCH FACILITY
FOR FUSION BLANKET/SHIELD TEST PROGRAM (TPE-II)

The Babcock & Wilcox Company has a thermal-hydraulic research facility which will be of interest to EG&G in the DOE-sponsored fusion program. The research facility is the Heat Transfer Facility (HTF). It was built in 1976 and is located at the Alliance Research Center in Alliance, Ohio. Brochures describing both the HTF and the R&D Division are enclosed to provide an overview of our facilities and capabilities.

Responses to the "Questionaire for TPE-II Facility Survey" regarding the HTF are provided below. The HTF can be used to accomplish both the thermal-hydraulic and thermomechanical testing. We do not have facilities to provide nuclear heating of the test specimens. The responses provided follow the format suggested by the questionaire

OVERVIEW TO HTF

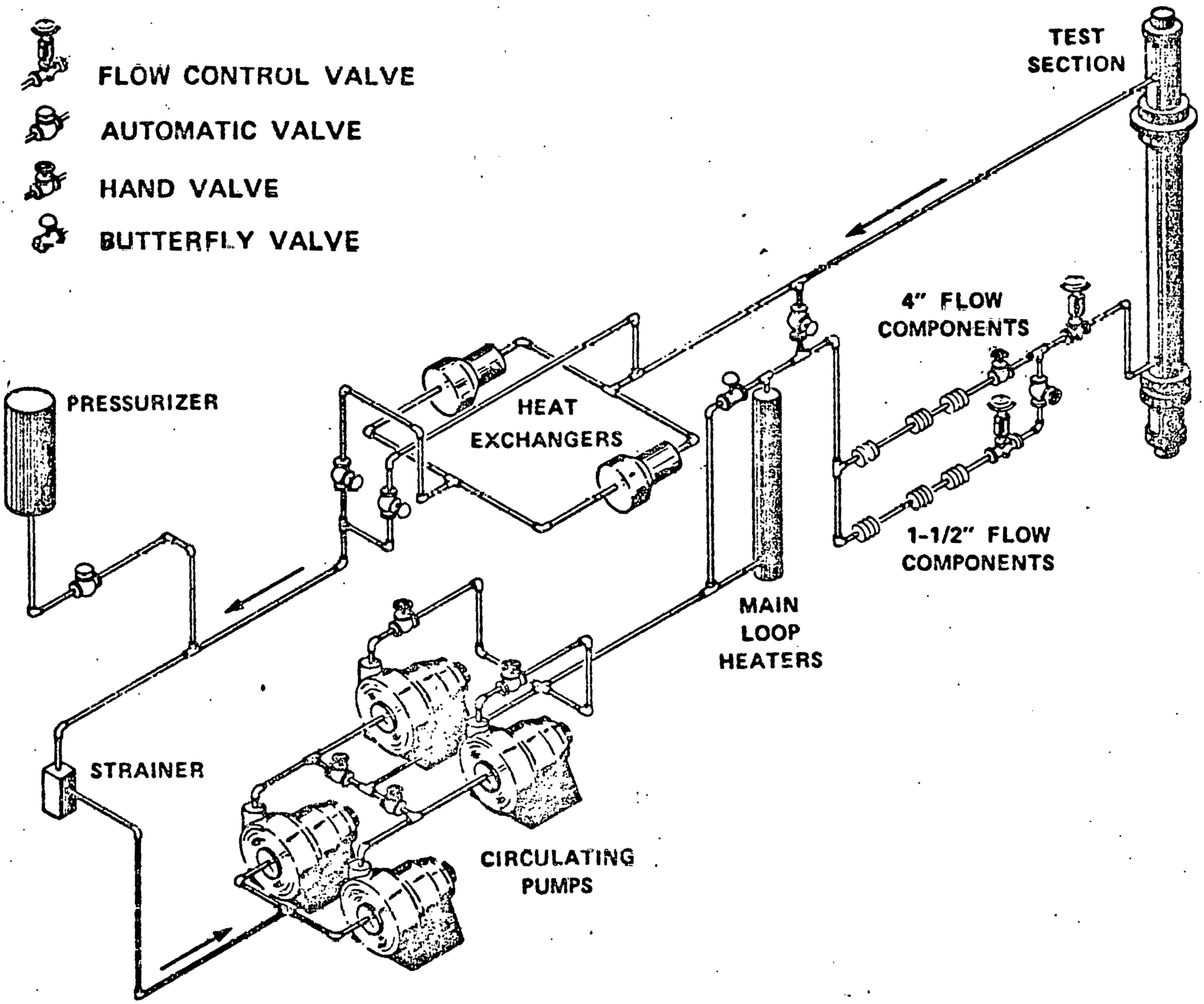
- Facility Configuration:
  The facility consists of
  - closed-circuit, two-phase pressurized water flow loop.
  - 10 megawatt power supply and a heat rejection system.
  - controls and instrumentation.
  - dedicated computer-based data acquisition system.

  The flow loop is shown schematically in Figure 1.

- Working Fluid: The working fluid is high purity water.

- Heat Source and Sink: The heat source for the facility is a 10 megawatt direct current power supply. The 50,000 amperes power supply consists of five silicon-controlled-rectifier power modules with the voltage potential variable from 0 to 200 VDC.

  Heat is removed from the pressurized water loop via two compact heat exchangers, a secondary water system and an external cooling tower. The nominal heat rejection rate is 6 megawatts, although rates approaching 10 megawatts have been achieved at high loop temperatures.



HEAT TRANSFER FACILITY LOOP PIPING ARRANGEMENT

FIGURE 1

- Summary of Capabilities

Electric Power Supply	10 Megawatts
Voltage	200 Volts DC Maximum
Current	50,000 Amps
Flow Loop:	
Design Pressure	21 Mpa (3000 psi)
Design Temperature	370°C (700°F)
Design Flow Rate	40 ℓ/Sec (650 GPM)
Design Head	256 m (840 Ft)
Number of Pumps	4
Number of Heat Exchangers	2 (6 Megawatts Nominal)
Material	Stainless Steel
Number of Test Bays	2

DETAILED DESCRIPTION OF HTF

Two of the most significant requirements of the fusion blanket/shield test program will be satisfied by the HTF. The first requirement is a source of pressurized water coolant with sufficient capacity to cool the test pieces. The second requirement is a source of electrical power for the heaters required to simulate the blanket heat release. The following paragraphs briefly describe the characteristics of the HTF in light of these two requirements. Several additional features of the facility that are important to the test effort are also described.

The HTF was designed with flow capabilities suitable for testing both nuclear reactor and fossil fuel boiler components. The facility has a closed-circuit flow loop that is constructed of stainless steel and designed to ASME specifications for a pressure of 21 Mpa and a temperature of 370^{o}C. The flow loop is shown in Figure 1. The volume of the piping was mimimized to gain fast response to thermal changes and to minimize the volume of steam that can exist. A bypass around the test section area provides added flexibility in control of conditions and response. There are four Chempump canned rotor pumps each having a head of approximately 130 meters at a design flow of 0.02 cubic meters per second. The pumps can be operated in any of the following conditions: one pump alone, two pumps in series, two pumps in parallel or two pairs of series connected pumps in parallel. This gives wide flexibility in coolant flow rates that range from 0.0006 cubic meters per second up to the maximum design point of 0.041 cubic meters per second at 260 meters head. Substantially more flow is obtainable at lower heads.

The circulating loop is connected to an ASME Section VIII code stamped pressurizer vessel. The pressurizer is used to control the loop pressure and to absorb the changes in volume of the loop fluid during operation.

Heat is removed from the pressurized water coolant through two compact heat exchangers in conjunction with an external cooling tower. The maximum heat rejection rate is determined by the loop temperature. The nominal rate is 6 megawatts, although rates approaching 10 megawatts have been achieved at high loop temperatures.

The loop can be controlled either manually or automatically through control valves, pressurizer heaters, loop heaters and a feed pump. The loop is also equipped with many interlocks and safety trip circuits to provide protection of the loop and test section.

A 10-megawatt power supply is available to simulate heat released in the blanket region. The 50,000-ampere power supply equipment consists of five silicon-controlled-rectifier (SCR) power modules capable of delivering up to 10,000 amperes each with the voltage potential variable from 0 to 200 VDC. The modules can be paralleled and controlled as a single package, or grouped for control of up to three packages simultaneously and/or individually. The control circuits can be readily extended to allow programmed computer control of any or all of the three packages.

A central control room contains all controls and recorders for complete control of the power equipment, the circulating loop and auxiliary support systems. It also contains the dedicated computer-based data acquisition system which has both a 192-channel high-speed scanner (up to 8000 channels per second) and a 200-channel low-speed (high resolution) scanner. The heart of the data acquisition system is a Varian V-75 computer. It can be programmed for simultaneous loop monitoring and protection, data acquisition, data reduction and test piece protection. All instruments and scanners feed through a wiring patch panel for simplified hook-up of instruments.

The loop instrumentation has additional features that are important to this program. All the instruments critical to the test have certified calibrations traceable to the NBS. Also, loop pressure, flow nozzle pressure drop, coolant flow rate, coolant temperature at the flow measuring components, coolant test vessel inlet and outlet temperature are all measured by more than one instrument. This allows for early detection of instrument failures and helps to establish measurement uncertainties by cross-checking.

There are several additional advantages of having the HTF available for use in the fusion blanket/shield test program. The first is that the facility is at its optimum age (4 years) for use. It is new enough that the most up-to-date systems have been incorporated into it and yet it is old enough that the systems have been debugged and proven in use. Another advantage is that many of the components used in the facility represent very long-lead-time items. If such a facility had to be constructed and debugged as part of this test program, it would have a significant impact on the time schedule. Another advantage is construction cost savings. No significant modifications to the facility are anticipated so facility construction costs should be minimal.

The HTF is comprised of two test bays and the necessary flow loop and power supplies. Testing at either site can be accomplished by the proper location of blank flanges. One test bay will be dedicated to the proposed program. This will permit the uninterrupted fabrication and assembly of the required test apparatus. The pressurized water flow loop, power supply, instrumentation and controls of the HTF will be shared with other tests being performed in the other bay. Scheduling conflicts are <u>not</u> anticipated. Should such conflicts arise during the execution of the program, B&W will quickly address the problem and attempt to minimize (if not preclude) any delay.

The cost for using the HTF will be based upon (1) a proration of the annual fixed costs associated with the facility and (2) the costs associated with operating the facility for the test program. The percentage of the fixed costs (which include annual depreciation charges and the labor, material and utility charges required to maintain the facility) will be prorated by the number of days/year which the facilit is dedicated to the fusion program. The facility will be considered dedicated to this program only during those days which the facility is in operation to either chec out the test apparatus or to execute the test program. The fixed costs for the HTF are <u>approximately</u> \$300,000 per year. The operating costs for using the facility include the labor, material and utilities (electric and water) required to perform the specific test program and to return the facility to its original condition upon completion of testing. These costs depend upon the scope of the test program and cannot be estimated at this time. Thus, the total cost for the use of the HTF during a given year will be calculated as follows:

$$\left\{\begin{array}{l}\text{total annual cost for}\\ \text{use of Heat Transfer}\\ \text{Facility}\end{array}\right\} =$$

$$\left\{\left(\begin{array}{l}\text{annual fixed costs}\\ \text{for the facility}\end{array}\right) \times \left(\frac{\text{number of days/year the facility is dedicated to the program}}{\text{365 days}}\right)\right\}$$

$$+ \left\{\begin{array}{l}\text{all operating costs (labor, materials, utilities, and indirects) and}\\ \text{applicable fee required to perform the specified test program}\\ \text{and return the facility to its original condition}\end{array}\right\}$$

AVAILABLE SUPPORT FACILITIES

The technical expertise and the research facilities available at the two research centers of Babcock & Wilcox are well qualified to perform both the thermal-hydraulic and thermomechanical test programs. The enclosed R&DD brochure outlines the capabilities of our research organizations.

The support facilities which appear to be needed for this fusion blanket/shield program are:

Experimental Stress Facilities

A summary of the facilities available for experimental stress analysis at the Alliance Research Center is as follows:

- Full range of equipment for installation, checkout and calibration of weldable high temperature strain gages and displacement transducers.
- All equipment necessary for performing the dynamic structural response tests - including accelerometers and mounts, cabling, signal conditioning, tape recorders, exciters (random and impulse) and signal sources.
- Data Acquisition System (small system) - All equipment necessary for assembly of a small data acquisition system including power supplies, signal conditioning, tape recording, signal monitoring, calibration equipment (NBS traceable).
- Data Acquisition System (large scale) - B&W has designed and built a 100+ channel data acquisition, monitoring and recording system. This system includes over 100 power supply/signal conditioning modules and ranging amplifiers compatible with strain gages or accelerometers, provision for monitoring quality of the incoming the recorded signals, frequency domain analog multiplexing, tape recording and built-in NBS traceable calibration facilities (signal sources, voltmeters etc.). This system also incorporates limited data reduction capability for reduction of real-time signals during testing.
- Data Analysis Laboratory - A fully equipped data analysis laboratory is available for in-depth data analyses (both time domain and frequency domain) using a combination of traditional laboratory measuring devices

and minicomputer-based analyzers. Traditional equipment is fully integrated into a laboratory patch panel, and includes voltmeters, oscilloscope, oscillograph, XY recorder, filters, analog signal manipulation modules and timer/counters.

Minicomputer based analyzers include a Hewlett-Packard 5451-C Fast Fourier Analyzer and a Gen Rad Series 2500 analyzer with time series and modal analysis software. Both machines are fully-expanded and up to date representing current state-of-the-art in experimental structural analysis equipment. Both machines are interfaced to a PDP 11/70 and a VAX 11/780 via a high speed data network, providing the capability for large scale data reduction tasks.

The data analysis system can provide complete and rapid reduction of all data from both the dynamic response testing and thermal effects testing.

Computer Facilities

At the Alliance Research Center, engineers and scientists have available the resources of a large computer, a Control Data Corporation CDC 7600 system capable of solving very large engineering programs. A data communications facility links this computer to a Digital Equipment Corporation VAX 11/780 system. This is an interactive scientific computer with which engineering models are defined, coded, debugged and tested. The resulting computer code is then transferred to the CDC computer for ultimate high-speed processing. Modifications and enhancements can be quickly checked on the VAX 11/780 computer using its interactive facilities and then incorporated into the model residing on the CDC computer. Similarly, data can be checked for errors, accuracy and completeness interactively using VAX 11/780 and thus, the final error free data can be transmitted to the CDC 7600 for batch computing services. When the results of this computer analysis are available, the information can be sent back to the VAX 11/780 for further analysis and output. For example, information may be displayed graphically, via CRT or plotter. The results can be stored locally for later comparison with results of other studies. All software and data formats on the computer system are compatible so that little engineering effort is required to format the data and code to transport it from one machine to the other. The Alliance Research Center provides a computer staff to provide support services for these computers. Services include system and applications consulting, data entry and computer operations support personnel.

The B&W Computer Center used by the Lynchburg Research Center (LRC) of the R&DD represents state-of-the-art capability in automated scientific computing designed to meet the expanding needs for computing requirements for the contract research and development programs. The system has been continuously enhanced to meet competitive standards in the computer services market place. The B&W Computer Center includes interconnected CDC Cyber 76, Cyber 73, and Cyber 171 computers complemented by a host of peripheral and communications equipment. The Cyber 73/ Cyber 171 process all interactive work and act as a system interface for all batch processing. The Center is supported by a number of experienced groups including software systems, communications, operation, and I/O control. In addition, there is also the B&W Hybrid Computer Laboratory which consists of three digital computers and two analog computers. These computers are inter-connected to allow two independent small hybrid computers or one large hybrid computer capable of simulating power generation systems from the heat source to the turbine-generator. The Laboratory consists of CDC-1700, EAI-640 and AP-120B digital computers interfaced with two fully expanded EAI-680 analog computers. There is a 1718 Satellite Computer between the CDC-1700 and the Cyber 73 of the B&W Computer Center. The Laboratory is supported by a number of experienced model developers, on-location maintenance and special hardware development personnel, and experienced systems and applications programmers.

Vacuum Chamber

The Alliance Research Center has a large vacuum chamber which is available for use in this test program. The chamber is a cylinder approximately 1 meter in diameter and 7 meters long. The chamber is exhibited in Figure 2. Each end has access through a hinged door. A door (approximately 1 meter in diameter) is also located at the middle of the cylinder and is equipped with an observation window. Also, a one-meter diameter flanged access in the vertical direction is available at the middle. The chamber is equipped with mechanical pumps, a diffusion pump and all the necessary valves. Test pressures of approximately 10^{-6} torr should be possible. Some maintenance will be required to put the chamber into service on this program. However, the costs involved should be minor compared to a new chamber and pumps. This represents a significant cost savings to the program and eliminates the long lead time associated with procurement of a suitable new chamber.

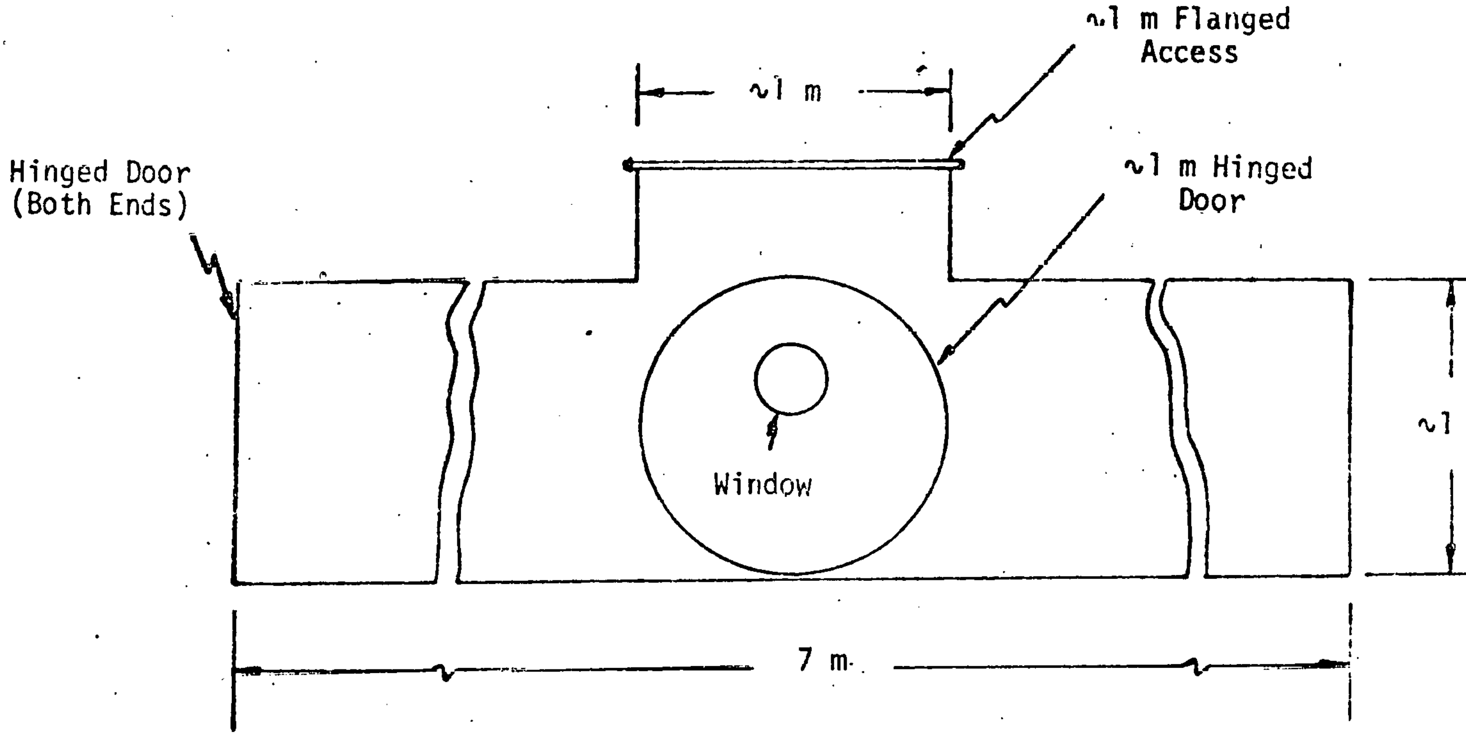


FIGURE 2. VACUUM CHAMBER

PROJECT TEAM

Babcock & Wilcox has both the managerial and technical capabilities to perform large experimental programs. The fusion blanket/shield test program would be performed under the technical management of the Thermal and Fluids Technology Section at the Alliance Research Center and would be administered by the Contract Research Division. Dr. P. A. Pfund is Manager of the Thermal and Fluids Technology Section. Mr. R. A. Kuchner, a Group Supervisor in that section, would be the Program Manager. Mr. M. Lukasik would be the contract administrator within the Contract Research Division.

A staff of highly qualified Support Group Leaders would assist the Program Manager in the performance of the technical tasks. The Support Group Leaders are

	Company Title	Program Position
K. E. Kneidel	Group Supervisor	Heat Transfer Support Leader
W. A. Fiveland	Research Specialist	Thermal-Hydraulic Research Specialist
S. E. Reed	Group Supervisor	Experimental Mechanics Support Leader
J. F. Martin	Sr. Design Engineer	Designs Support Leader
R. J. Lepucki	Instrument Engineer III	Instrument and Controls Support Leader

The resumes of Dr. Pfund, Mr. Kuchner, Mr. Lukasik and each support group leader follows.

Resume of PHILIP A. PFUND, Section Manager, Thermal & Fluids Technology Section
Alliance Research Center, Alliance, Ohio

Education:

University of Wisconsin, Madison, Wisconsin, B.S. Applied Mechanics and Engineering Physics, 1966
University of Minnesota, Minneapolis, Minnesota, M.S. Fluid Mechanics, 1969
University of Minnesota, Minneapolis, Minnesota, Ph.D. Fluid Mechanics, 1975

Professional Experience:

(1980-Present) THE BABCOCK & WILCOX COMPANY, RESEARCH AND DEVELOPMENT DIVISION, SECTION MANAGER, THERMAL & FLUIDS TECHNOLOGY. Managerial responsibility for engineers and projects in the areas of heat transfer, fluid mechanics and thermal systems. Section responsibilities include: large-scale heat transfer experiments, reactor core thermal performance, boiler cleaning equipment, heat transfer analysis and computer code development, thermal insulation, reactor core hydraulic performance, control valve performance, laser velocimetry, flow modeling, fluid metering, critical heat flux testing, two-phase flow heat transfer, basic thermal radiation properties.

(1977-1980) GROUP SUPERVISOR, THERMAL & FLUIDS TECHNOLOGY . Supervisory responsibility for projects dealing with nuclear reactor vessel model flow testing, laser doppler velocity measurements of liquids, solids, and two-phase flows, nuclear fuel assembly hydraulics, molten steel hydraulics, fluidized bed heat treating.

(1975-1977) SENIOR RESEARCH ENGINEER, APPLIED MECHANICS. Project responsibility for control rod drive mechanism testing, development and testing of electrical insulation systems for control rod drive motors.

(1973-1975) RESEARCH ENGINEER, SPECIAL PROJECTS. Project responsibility for mechanical and hydraulic testing of nuclear fuel assembly designs for maritime application, hydraulic model testing of maritime reactor internals.

Professional Affiliations:

American Society of Mechanical Engineers, 1973-Present
ASME, Fluid Mechanics Committee, 1976-Present
ASME, Canton-Alliance-Massillon Section Board of Directors, 1977-Present
IEEE, Electrical Insulation Group, 1975-1977

Publications:

P. A. Pfund, "Heat Transfer from the Inner Wall of an Annulus to a Decaying Swirling Flow", M.S. Thesis, University of Minnesota, 1969.

PHILIP A. PFUND

P. A. Pfund, "Heat Transfer and Fluid Flow in a Short Vortex Chamber", Ph.D. Thesis, University of Minnesota, 1975.

P. A. Pfund, D. C. North and R. J. Parekh, "Evaluation of a Fuel Assembly for a Nuclear Merchant Ship", ASME Paper 76-WA/NE-18, 1976.

S. C. Yao and P. A. Pfund, eds., "Fluid Flow and Heat Transfer Over Rod or Tube Bundles", ASME Symposium Volume G00157, December 2-7, 1979.

Resume of MICHAEL LUKASIK, Senior Contract Manager
Contract Research Division, Alliance, Ohio

Education:

Broome Technical Community College, Binghamton, New York
A.A.S. Mechanical Tech., 1966
University of Cincinnati, Cincinnati, Ohio
Part-time M.E. study, 1967-1970
University of Akron, Akron, Ohio
B.S. Technical Education, 1972
University of Akron, Akron, Ohio
M.S. Technical Supervision and Administration, 1974

Professional Experience:

(1966-1970) Engineering Assistant positions with the General Electric Co. and Garrett Corporation, with primary effort centering on developmental testing of gas turbine engines.

(1970-Present) BABCOCK & WILCOX COMPANY, as follows:

1970-1972 Engineer in Nuclear Equipment Division, Barberton, Ohio. Responsible for design and stress analysis of heavy lifting equipment associated with pressure vessel fabrication.

1972-1974 Senior Stress Engineer in Nuclear Equipment Division, Barberton, Ohio. Responsible for stress analysis, Occupational Safety & Health Act (OSHA) lifting compliance, and conformance to contractual requirements on pressure vessel handling.

1974-1976 Contract Administrator in Nuclear Equipment Division, Barberton, Ohio. Responsible for all administrative matters associated with a Naval Nuclear Reactor Vessel contract.

1976-1980 Contract Manager in the Contract Research Division, Alliance, Ohio. Responsible for the management of a variety of energy research and development contracts.

1980-Present Senior Contract Manager in the Contract Research Division, Alliance, Ohio, with continued primary management functions on selected contracts in addition to supervisory responsibilities for contract manager group.

Professional Affiliations:

Part-time Instructor at the University of Akron, including Statics, Strength of Materials, Kinematics and Management as courses instructed.

Licenses:

Registered Professional Engineer, Ohio, 1976.

12/17/80

Resume of ROBERT A. KUCHNER, Group Supervisor, Thermal & Fluids Technology Section
Alliance Research Center, Alliance, Ohio

Education

Newark College of Engineering, BS (Mechanical Engineering), 1969
Pennsylvania State University, MS (Mechanical Engineering) 1979

Professional Experience

THE BABCOCK & WILCOX COMPANY, RESEARCH AND DEVELOPMENT DIVISION

(Present) GROUP SUPERVISOR, THERMAL & FLUIDS TECHNOLOGY. Supervises a group of engineers, technologist and technician responsible for performing large-scale heat transfer experiments. Experiments include critical heat flux testing for nuclear reactor fuel pin assemblies and life testing of boiler membrane wall panels subjected to water lancing. Facilities include a $3 million, 10 MW two-phase heat transfer research facility.

(1979-1980) SENIOR RESEARCH ENGINEER, THERMAL & FLUIDS TECHNOLOGY. Directed a multidisciplined team of engineers investigating the effect of water lancing of fireside deposits in coal-fired utility boilers on the life of boiler tubes. This large, 3-year, contract-sponsor research program included analytical modeling of the thermal, stress and fatigue crack growth processes, a laboratory experiment and a field test program.

(1974-1979) RESEARCH ENGINEER, REACTOR TECHNOLOGY. Provided general heat transfer consulting services including:

-- performing multidimensional conduction heat transfer analyses.
-- measuring thermophysical properties of nuclear grade alloys and reflective, all-metallic insulation.
-- reviewing new technologies such as heat pipes, laser pulse diffusivity measurement techniques and water cleaning of fossil boilers.
-- performing thermal energy surveys.

Selected and implemented advanced finite difference programs for general purpose, multidimensional conduction heat transfer analyses.

Analyzed, designed and numerically simulated major components for a 10 MW, 700 GPM, 3000 psi, two-phase heat transfer research facility.

(1973-1974) PENNSYLVANIA STATE UNIVERSITY, MECHANICAL ENGINEERING DEPARTMENT, GRADUATE ASSISTANT.

(1969-1973) UNITED STATES AIR FORCE, ELLSWORTH AIR FORCE BASE, CAPTAIN, USAF.

CHIEF, OPERATIONS AND MAINTENANCE: Supervised a work force of 420 civilian and military personnel engaged in the maintenance, repair and alteration of base facilities, utility systems, missile site facilities and airdrome. The estimated cost of these facilities and systems was $260 million. Annual operating budget for this system was $4.4 million.

MECHANICAL ENGINEER: Designed heating, ventilating, air conditioning and fire protection systems for base facilities. Supervised Architect-Engineer design projects, including the design of a $1.2 million commissary store and warehouse. Coordinated the $1.1 million base utility program. Provided engineering consulting services for central heating plants, natural gas distribution system, data processing center and composite medical facility.

Professional Affiliations:

Professional Engineer: Ohio, 1976
American Society of Mechanical Engineers, Member
Tau Beta Pi - National Engineering Honor Society
Pi Tau Sigma - National Mechanical Engineering Honor Society
Who's Who in American Colleges and Universities, 1969

Resume of KURT E. KNEIDEL, Group Supervisor - Heat Transfer, Thermal & Fluids Technology Section, Alliance Research Center, Alliance, Ohio.

Education:

Pennsylvania State University, BS (Mechanical Engineering), 1970
The University of Akron, MS (Mechanical Engineering), 1973

Professional Experience:

(1981-Present) THE BABCOCK & WILCOX COMPANY, RESEARCH AND DEVELOPMENT, GROUP SUPERVISOR - HEAT TRANSFER, THERMAL & FLUIDS TECHNOLOGY. Supervises a group engaged in analytical and experimental heat transfer work. Activities include testing and analysis of homogeneous insulations and MIRROR all-metal reflective insulation for nuclear service, design and evaluation of high temperature heat exchangers, developing basic computational tools for radiation, conduction and free convection, and providing general technical assistance in the solution of thermodynamic and heat transfer problems.

(1980-1981) THE BABCOCK & WILCOX COMPANY, RESEARCH AND DEVELOPMENT DIVISION, RESEARCH SPECIALIST, THERMAL & FLUIDS TECHNOLOGY. Responsible for critical heat flux testing technology and applications of the Babcock & Wilcox Heat Transfer Facility.

(1977-1980) SENIOR RESEARCH ENGINEER, THERMAL & FLUIDS TECHNOLOGY. Responsible for design, construction, testing and analysis of a series of six large-scale simulations of PWR fuel assemblies. This was a 4-year effort funded at approximately 3 million dollars. These critical heat flux tests involved pressurized water at a wide range of conditions and electrically-heated test sections with heat fluxes up to 3 Mw/m^2. All testing was conducted on the Heat Transfer Facility. Project responsibilities included coordination and guidance of personnel in design, computer programming, instrument development, purchasing, metallurgy, welding, corrosion and test facility operation.

Co-developer of a critical heat flux detection system for use in nonuniformly heated tube bundles based on ultrasonic thermometry. Project responsibilities included sensor design, optimization of the system for use on the Heat Transfer Facility and supervision of computer programming and instrumentation personnel.

(1973-1977) RESEARCH ENGINEER, REACTOR TECHNOLOGY. Responsible for design, construction, critical heat flux testing and analysis of electrically-heated rod bundle simulations of PWR fuel assemblies. Testing was done on the 2 megawatt Boiling Heat Transfer Facility. Conducted thermal analysis and testing of heat transfer systems involving conduction, convection and radiation heat transfer.

(1971-1973) SENIOR ENGINEER, REACTOR TECHNOLOGY. Conducted thermal analysis and mathematical model development for Mirror reflective insulation.

(1970-1971) PRATT AND WHITNEY AIRCRAFT, FLORIDA RESEARCH AND DEVELOPMENT CENTER, ENGINEER, APPLIED RESEARCH. Conducted analytical design work for cooling both air breathing and non-air breathing engine exhaust nozzles under high heat flux.

Professional Activities:

Registered Professional Mechanical Engineer in Ohio
American Society of Mechanical Engineers, Member
Tau Beta Pi - Honorary Society in Engineering, Member
Pi Tau Sigma - Honorary Society in Mechanical Engineering, Member
Phi Kappa Phi - Honorary Society (General Scholastic), Member

Presentations and Publications:

James E. Smith and Kurt E. Kneidel, "Conservation of Energy Through Advanced Thermal Insulation Design", Presented at the MIRROR Insulation Inservice Inspection Symposium, December, 1973.

K. E. Kneidel and P. M. Gerhart, "Further Computations of Supersonic Free Shear Layers", Journal of Fluids Engineering, Vol. 96, No. 4, p. 401, 1974.

A. R. Barber, K. E. Kneidel, C. S. Fitzgerald, L. C. Lynnworth, "Ultrasonic Temperature Profiling System for Detecting Critical Heat Flux in Nonuniformly Heated Tube Bundles", Journal of Heat Transfer, Vol. 101, No. 4, p. 622, 1979.

Resume of WOODROW A. FIVELAND, Research Specialist, Thermal & Fluids Technology, Alliance Research Center, Alliance, Ohio

Education:

University of Nebraska, BS (Mechanical Engineering), 1971
University of Nebraska, MS (Mechanical Engineering), 1972
Thesis: Unsteady Channel Flow Between Porous Plates, Department of Mechanical Engineering, Lincoln, Nebraska, 1972
University of Akron, PhD (Mechanical Engineering) 1978
Thesis: An Analytical Study of Laminar and Turbulent Magneto-Fluid Dynamic Boundary Layer Flow with Heat Transfer Department of Mechanical Engineering, Akron, Ohio, 1978

Professional Experience:

(1980-Present) THE BABCOCK & WILCOX COMPANY, RESEARCH AND DEVELOPMENT, RESEARCH SPECIALIST, THERMAL & FLUIDS TECHNOLOGY. Responsible for formulating basic models of insulation heat transfer including the effects of conduction, convection and radiation. Radiation heat transfer models were designed to simulate simultaneous absorption, scattering and remission of energy. Involved in developing Mie scattering computer programs for cylindrical and spherical bodies. Responsible for formulating various basic computer codes which solve problems of radiation heat transfer, and melting and solidification.

(1977-1980) UNIVERSITY OF AKRON, AKRON, OHIO, ADJUNCT PROFESSOR. Responsible for teaching courses in heat transfer and numerical analysis.

(1977-1980) THE BABCOCK & WILCOX COMPANY, RESEARCH AND DEVELOPMENT, SENIOR RESEARCH ENGINEER, THERMAL & FLUIDS TECHNOLOGY. Responsible for formulating basic insulation heat transfer models including the effects of conduction, convection and radiation. Responsible for formulating basic computer codes which solve problems of radiation heat transfer and melting and solidification.

(1976-1977) Educational leave at the University of Akron. Consultant to The Babcock & Wilcox Company, Research and Development, Reactor Technology.

Conducted a study of fusion reactors and the connected engineering design difficulties with them. Collaborated in the development of a computer program to monitor a high temperature, high pressure boiling heat transfer facility.

(1974-1976) THE BABCOCK & WILCOX COMPANY, RESEARCH AND DEVELOPMENT, RESEARCH ENGINEER, REACTOR TECHNOLOGY. Experimentally determined critical heat flux in simulated nuclear fuel assemblies. Conducted testing and analysis of rupture tests to benchmark nuclear fuel pin rupture during a loss-of-coolant accident.

(1972-1974) THE BABCOCK & WILCOX COMPANY, RESEARCH AND DEVELOPMENT, SENIOR ENGINEER, REACTOR TECHNOLOGY. Conducted experiments of critical heat flux which characterized the thermal performance of nuclear fuel assemblies. Designed tests to characterize the rupture of nuclear fuel pins during a postulated loss-of-coolant accident. Developed a fast-response, high-temperature radiation heater.

Professional Activities: Member

Member of ASME (K-8) technical committee on Theory and Fundamental Heat Transfer Research.

Coorganizer and cochairman of a session on Heat Transfer in Composites at ASME Winter Annual Meeting, New York, December 1979.

Professional Affiliations:

American Society of Mechanical Engineers, Member
Sigma Tau Honorary Society in Engineering, Member
Tau Beta Pi Honorary Society in Engineering, Member
Pi Mu Epsilon Honorary Society in Mathematics, Member
Sigma Xi Research Honorary Society, Member

Presentations and Publications:

W. A. Fiveland, A. R. Barber and A. L. Lowe, Jr., "Rupture Characteristics of Zircaloy-4 Cladding with Internal and External Simulation of Reactor Heating", Presented at the Third International Conference on Zirconium in the Nuclear Industry, Quebec City, Quebec, 1976 and published in Zirconium in the Nuclear Industry, ASTM-STP-633, pp. 36-49, 1977.

W. A. Fiveland and P. C. Lu, "Transient Boundary Layers Between Porous Plates", Presented at the Winter Annual Meeting, New York, New York, 1976 and published in Journal of Applied Mechanics, Vol. 98, No. 4, 555-558, 1976.

W. A. Fiveland, "An Analytical Study of Laminar and Turbulent Magneto-Fluid Dynamic Boundary Layer Flow with Heat Transfer", University Microfilms, 1978.

W. A. Fiveland and B. T. F. Chung, "Unsteady Heat Transfer for Laminar MFD Flow Over a Flat Plate",Paper No. 79WA/HT-25, Presented at the Winter Annual Meeting, New York, 1979.

W. A. Fiveland and B. T. F. Chung, "Numerical Solution for Heat Transfer in Turbulent MFD Boundary Layer Flow", Numerical Methods for Non-Linear Problems - Proceedings of the International Conference held at University College Swansea U.K., Volume 1, 1980.

(Others furnished upon request)

Resume of STUART E. REED, Group Supervisor, Applied Mechanics Section
Research and Development Division, Alliance, Ohio

EDUCATION

Bucknell University, Lewisburg, Pennsylvania, BS Mechanical Engineering, 1970
Bucknell University, Lewisburg, Pennsylvania, MS Mechanical Engineering, 1970

PROFESSIONAL EXPERIENCE

(1970-Present) BABCOCK & WILCOX, RESEARCH AND DEVELOPMENT DIVISION, Alliance, Ohio, GROUP SUPERVISOR. Since joining Babcock & Wilcox in 1970 as a senior engineer, responsibilities have included numerous projects in the experimental mechanics area, both static and dynamic. Extensive laboratory and field experience with strain gages and accelerometers at ambient and elevated temperatures, as well as photoelastic stress analysis and seismic qualification. Currently group supervisor of the Experimental Mechanics Group, with responsibility for experimental stress analysis, special transducer design, and residual stress measurement by both x-ray diffraction and hole-drilling techniques.

PROFESSIONAL AFFILIATIONS

American Society of Mechanical Engineers (1970-Present)

Society for Experimental Stress Analysis (1971-Present)

SESA, Technical Committee on Strain Gages (1979-Present)

PUBLICATIONS

Static and Dynamic Analysis of Reactor Core Support Structures, coauthor, American Nuclear Society Annual Meeting, Los Vegas, Nevada (June 18-22, 1972).

Analytical and Experimental Stress Analysis of a Cylinder-to-Cylinder Structure, coauthor, presented at 4th International Conference on Structural Mechanics in Reactor Technology, San Francisco, California (August 15-19, 1977), and at the Energy Technology Conference and Exhibition, Houston, Texas (September 18-22, 1977), ASME Paper No. 77-PVP-11.

LICENSES

Professional Engineer, Ohio

PATENTS

U.S. #4208567, "Weldable Instrumentation Installation Tool".

1/1/81

Resume of JEFFREY F. MARTIN, Senior Design Engineer, Design Engineering Section, B&W Research & Development Division, Alliance, Ohio 44601

Education:

Fenn College of Engineering of Cleveland State University, Cleveland, Ohio, BME (Co-op) 1970.

Professional Fxperience:

(1966-1969) STUDENT ENGINEER on Co-operative Program.

(Two quarters) MORGAN ENGINEERING COMPANY, Alliance, Ohio, DRAFTSMAN.

(Four quarters) BABCOCK & WILCOX COMPANY, POWER GENERATION DIVISION, Barberton, Ohio, DESIGN DRAFTSMAN, Plant Engineering.

(One quarter) BABCOCK & WILCOX COMPANY, POWER GENERATION DIVISION, Barberton, Ohio, SUPERVISORY ASSISTANT, Maintenance and Construction.

(1970-1974) BABCOCK & WILCOX COMPANY, Alliance Research Center, Alliance, Ohio, TEST ENGINEER for mechanical testing of Maritime and Commercial Nuclear Reactor Components. These included hydraulic control rod drive, roller nut control rod drive, fuel assembly life and vibration testing and fuel assembly mechanical characterization tests.

(1974 - Present) DESIGN ENGINEERING SECTION, SENIOR DESIGN ENGINEER. Responsible for the design of various test facilities including high pressure piping systems and pressure vessels. Responsibiliites include extensive use of the ASME Boiler and Pressure Vessel Codes and ANSI B.31.1 Power Piping Code.

Licenses:

1. Engineer in training, Ohio
2. Division of Pressure Piping, Ohio, Special Inspector of Pressure Piping

Resume of RICHARD J. LEPUCKI, Instrument Section, B&W Research & Development Division, Alliance, Ohio

Education:

Stevens Institute of Technology, BE (Electrical Engineering), 1971
University of Akron, Master's Program (Electrical Engineering) 1974-1976.

Professional Experience:

BABCOCK & WILCOX COMPANY, Research & Development Division

(1978 - Present) INSTRUMENT ENGINEER III, INSTRUMENT SECTION. I am responsible for the design, development of measurement and control systems and related electrical and electronic equipment. Areas covered include analog circuitry, digital circuitry, process control system analysis and design, industrial electrical power systems and equipment, measurement technology and data acquisition systems.

I have designed the flow control and measurement system and remote data acquisition system for a flow restrictor development test program at a TVA power plant. I have designed the electric heater, flow, and cycling controls for a test of water lancing of fireside deposits of utility boilers. I have participated in the start-up of a large heat treat furnace and other heat treat facilities for a manufacturing division. I have reviewed high energy spark ignitor power supplies used for igniting heavy fuel oils.

(1971-1978) INSTRUMENT ENGINEER, DESIGN ENGINEERING SECTION. I was responsible for the instrumentation and controls and electrical phases for a multi-million dollar heat transfer facility. This included a 50,000 amp, 10 megawatt power supply system; flow, temperature and pressure controls; and a 400 channel computerized data acquisition system. My other work covered a broad variety of similar applications.

E.I. DuPONT DeNEMOURS & COMPANY

(1970) ENGINEERING AIDE, INSTRUMENT GROUP. I performed a study of measurement requirements and capabilities of in-plant equipment.

Professional Affiliations:

Institute of Electrical & Electronics Engineers

Brochures on the Heat Transfer Facility and B&W Research and Development are not included since they contain many color photographs which could not be reproduced for this report. These brochures may be obtained from EG&G Idaho Division of Fusion Technology.

APPENDIX H

APPENDIX H

COLUMBIA UNIVERSITY RESPONSE

Columbia University in the City of New York | *New York. N.Y. 10027*

CHEMICAL ENGINEERING·RESEARCH LABORATORIES
OF THE SCHOOL OF ENGINEERING & APPLIED SCIENCE

632 West 125th Street
~~UNiversity 5-8400~~
212-280-4163

September 25, 1981

Mr. P. Y. Hsu
TPE-11 Program Manager
Fusion Technology Program
EG&G Idaho, Inc.
P.O. Box 1625
Idaho Falls, Idaho 83415

Dear Mr. Hsu:

This is in response to your letter of September 11th to Professor R.A. Gross of the Department of Applied Physics and Nuclear Engineering at Columbia University.

Enclosed are responses to the questions of enclosure 2, together with a complete set of specifications and drawings for Columbia's thermal hydraulic test facility. We believe the facility is uniquely suited for the experimental work described, both in the facilities and engineering personnel available for design of the special test pieces required.

Should there be any questions, please call or write the facility whose telephone and address are above.

Very truly yours,

W.E. Hovemeyer

W.E. Hovemeyer, Manager
Heat Transfer Laboratory

WEH/bf

Encls.

QUESTIONNAIRE FOR TPE-II FACILITY SURVEY

1. WHAT FACILITIES DO YOU HAVE FOR CONDUCTING STEADY OR CYCLIC THERMO-MECHANICAL HEATING EXPERIMENTS?

a. What is the heat source?

The heat source is electrical energy. The electrical heating source is ripple free DC current from six DC generators. The steady state power available is 11.5 megawatts; 230 volts at 50,000 amperes. Short term transients in the millisecond region of duration are available and in excess of 25 megawatts. Intermediate duration, controlled transients are also available.

b. What is the test environment (vacuum, inert gas, etc.)?

Two high pressure, fully instrumented water flow loops are available for testing. The test environment is H_2O or D_2O at pressures up to 2400 psi and temperatures of 650° F. Test conditions and geometries can be altered to accommodate other environments, i.e., vacuum or inert gas.

c. What size and configuration of test pieces will they accommodate?

The sizes and configuration of test pieces are unlimited. Provisions, however, are required for heating by direct, electrical resistance heating or indirect electrical heating by thermal conduction or radiation. An engineering design staff is available for specific applications to accommodate insertion in Columbia University's thermal test facility.

d. What materials can be accommodated in the test space?

Any material can be accommodated in the test space. Many tests have been conducted on water cooled tubing for reactor studies. The tubing has been stainless steel, aluminum or inconel, however any water cooled geometry can be accommodated for testing.

e. What range of temperatures and/or power can be achieved?

Temperature ranges possible are limited only by temperature limits of material. Water cooled geometries can be tested in a water loop with temperatures 70° - 650° F and pressures up to 2500 psia.

The power is DC, ripple free electrical current:
Steady State: 11.5 megawatts - 230 volts, 50000 amperes, DC
Controlled Transient, medium duration - 15 megawatts 1-5 seconds
Short Time Transient - 30 megawatts - millisecond pulse.

1.f. What are the rise and decay times of temperature/power in these facilities?

Temperature rise and decay in a test material is a function of test piece mass, geometry and power transients available.

Power Transients

Rise Time - 10×10^{-3} seconds

Decay Time - 10×10^{-3} seconds

As mentioned above the rise and decay times can be lengthened and controlled by input programming the DC generator system.

g. What capability exists for active cooling of a test piece in these facilities?

Test pieces are cooled by pumping water past them. The water is cooled by an automated system of shell and tube heat exchangers and a bypass line. The secondary system is supplied by three pumps having a capacity of 800 gpm. The secondary-side coolant is well water at an average temperature of 64° F. The wells are located on-site.

h. How are these facilities instrumented?

The high pressure water flow loops are fully instrumented. All pertinent power, flow, temperature and pressure readings are collected and recorded on magnetic tape. All vital parameters are measured by at least two independent instruments. A computer controlled data acquisition system scans 120 channels, records data on magnetic tape, and performs on line data reduction. This system has high speed capabilities and has been used for transient work.

Transducers available are as follows:

1. Pressure - water or gas - 0 - 3000 psia
2. Differential pressure - 0 - 200 psid, 3000# gage
3. Turbine flow meters - 0 - 500 GPM, water
4. Sheathed, thermocouples, many types and capable of measurement to 2000° F at 3000 psia
5. Infra-Red temperature analyzer
6. Venturi flow meters - 0 - 500 GPM
7. Electrical system, voltage, current, power

All transducers have electrical outputs which are connected to shielded wires from any test area. The shielded, 1.h. Continued

low level systems are grounded to provide noise-free signal transmission. Instrumentation subject to high common mode voltage are isolated by means of amplifiers having common mode isolation of 700 volts. All transducer circuits are provided with external reference calibration in the main instrument room center, where complete programming of all instrumentation is available. Signal outputs are multiplexed by a high speed multiplexer rated at 45 KH_Z multiplex frequency and enabling use of multiplexer for transient studies, all multiplexed signals are connected to a 2100 A, Hewlett Packard on line computer. Teletype terminals for computer operation are available at all test locations. CRT readouts are located in the computer room. On line data reduction of variables and computed data can be made on demand from the test location. All experimental data is recorded on magnetic tape.

Complete instrumentation and diagrams are shown in the attached specification sheets.

i. <u>What is the availability and approximate operating cost of these facilities?</u>

The Heat Transfer Research Facility (HTRF) is part of Columbia University, and is therefore run on a non-profit basis. Direct cost plus overhead is charged to each sponsor. This non-profit status, plus a small full-time staff is responsible for generally keeping costs below what they are at other laboratories for similar work.

The HTRF is available for use by all appropriate sponsors. Test programs are scheduled according to work commitments; these schedules are taken seriously, and are usually met.

j. <u>What supporting facilities are available, e.g., machine shops, analytical laboratories, etc.?</u>

Complete machine shops are available at the facility and at Columbia University's main campus. The machine shops are capable of precision machine work. The facility has available complete chemical analytical laboratories and metals laboratories including stress analysis, failure regimes of metals, vibration and metal fatigue using acoustical techniques. The facility also houses electronic shops, computer center, photographic shop and welding shop.

1.k. What are the numbers and qualifications of personnel available to support tests in these facilities?

Personnel Available at Facility

5 Staff Engineers, Mechanical, Electrical, Instrumentation Computer, Nuclear and Project

2 Instrumentation Technicians

5 Mechanical Technicians and Shop Personnel

5 Administrative Supporting Staff Members

Qualified consultants in the areas of Chemistry, Nuclear, materials and metals, structural analysis and applied physics are available from the staff of Columbia University.

All above personnel have an average of 20 years experience.

l. What other aspects of these facilities would be pertinent to tests of this type?

A background of 30 years in the heat transfer field together with the most complete data in the country on fuel rod tests as applied to nuclear reactor cores to determine critical heat flux in reactor cores under operating conditions and also failure modes. The above data includes blow down and transient data which was done extensively at Columbia University.

2. WITH RESPECT TO THERMAL-HYDRAULIC TEST FACILITIES, PLEASE ANSWER (a) THROUGH (l) ABOVE AND THE FOLLOWING ADDITIONAL QUESTIONS.

a. What is the configuration of the facility (circulation loop, blowdown, two-phase, etc.)?

Two high pressure flow loops capable of blowdown and two-phase operation are available. The attached notes contain specifications and schematic drawings of these loops. Construction is of such a nature as to enable easy and rapid alterations, when necessary.

b. What fluids can be used? (H_2O, gas, liquid metal, etc.)

The two high pressure loops presently use H_2O or D_2O as the working fluid. However, alterations can be made to enable the use of inert gasses or vacuum.

c. What heat sources and sinks are available?

The heat source available is electric current provided

2.c. Continued

by six DC generators. A maximum of 11.5 MW is produced, comprised of 230 volts and 50,000 amps.

The heat sink is well water pumped from on-site wells through an automated heat exchange system. Presently 800 gpm of well water at 64°F is available for cooling. Additional pumping capacity can increase this flow limit, if necessary.

d. <u>What flow rates, pressures, pressure drops, velocities, etc., can be attained?</u>

These parameters are presently as follows:

Maximum pressure, psia	2400
Maximum test section flow, gpm	850
Head developed by pump, ft. of water	600

However, alterations in piping will result in different flow rates and velocities, as required.

3. <u>WHAT CAPABILITY DO YOU HAVE FOR PROVIDING NUCLEAR HEATING?</u>

There are no nuclear heating facilities available for this termal hydraulic facility.

Section 1

EXECUTIVE SUMMARY

The optimizing of nuclear power reactor design for efficiency of normal operation and for safety against abnormal or extreme contingencies requires a thorough and accurate understanding of the heat removal mechanisms and capabilities in the reactor core. This is the area of nuclear thermal-hydraulics.

The Heat Transfer Research Facility was founded by the AEC and duPont at Columbia University in 1950 to perform work in the nuclear thermal-hydraulic area. Placed in the current Chemical Engineering Research Station, it first tackled thermal-hydraulic questions for the low pressure Savannah River production reactors, then an AEC-Canada cooperative heavy water power program. Ever since, it has remained in the support of U.S. and foreign governments and industry.

This specific program supported by the Electric Power Research Institute will assist the Facility, which as a non-profit institution cannot accumulate substantial funds for new equipment from operating charges, to expand its thermal power and thermal-hydraulic capabilities. This will permit studies of increased heat generation densities, open core structures, hypothetical malfunctions, larger core simulations, etc., to improve still more the accuracy with which design optimizations can be carried out. The following sections of this summary deal sequentially with the numbered sections of this report.

The safety and operational limits of a nuclear power reactor are strongly influenced by the steady state and transient thermal-hydraulic behavior of the heat generating core in the reactor.

With the increased utilization of nuclear reactors for power generation, the need for extensive and reliable understanding of reactor thermal-hydraulics assumes a prominent role. The areas in which design and research information are needed for establishing the operational and safety limits are:

a. Critical heat flux limitations under steady-state and transient conditions over a wide range of process parameters for operating and postulated accident conditions.

b. Inter-fuel assembly mixing studies of large open-lattice PWR cores.

c. Steady-state and transient behavior of large open-lattice PWR cores.

d. Improved detailed measurement of local fluid conditions in rod arrays.

e. Transient two-phase flow and heat transfer under loss-of-coolant and loss-of-flow conditions.

f. Heat transfer and design evaluation of emergency core cooling systems.

Most of these experiments are intended to provide design data as well as to substantiate and validate the theoretical models used in reactor analyses.

There exist two sets of motor-generator (MG) sets at the Facility, one located in the basement, the other in the driveway. The original basement MG set consists of a single 13.8 kV, 5,000 HP A.C. motor connected to two 1750 kW, 175 V, 10,000 ampere D.C. Generators, and a 40 kW exciter for the motor and a 10 kW exciter each for the generators. The four driveway sets all have 13.2 kV, 2100 HP A.C. motors connected to 1500 kW, 250 V, 6,000 ampere D.C. Generators.

Start-up of the electrical system requires a procedure of 35 steps. Necessary primary and secondary protections are provided in the operating circuits.

The interconnections of the MG sets may be varied to obtain various maximum powers for the test sections. The original schedule of interconnections of the 6 generators is given in Table 1.1 herewith.

While this schedule could be followed at present, the cooling time back to a starting cold condition would be very substantial, and a rapid schedule of runs could not be maintained. Therefore. subsequently it was decided to acquire a variable tap transformer and full-wave 55 V D.C. rectifier to boost the 175 V output of the original (basement) generators to 230 V, so that it can be paralleled with the driveway generators. This utilizes the maximum voltage and current of all generators; the arrangement is described in Section 4. Subsequently, it was decided to down-rate the system for greater reliability and safety. The maximum D.C. test power obtainable on this basis remain about the same as in Table 1.1

Although the root mean square ripple of the transformer and rectifier are about 4.2% of the rectified voltage, a much smaller RMS fluctuation is obtained in actual use. The maximum it can reach is 4.2% of 55 volts at 11,000 amperes in series with 175 volts of D.C., while in parallel with 9,000 amperes without fluctuation. This yields roughly an overall ripple of 0.7%. At either lower voltages or lower currents it reduces further.

A maximum voltage of 180 V D.C. with currents of 58,000 amperes (10.4 MW) can be obtained without perceptible ripple, or to 240 volts D.C. and 47,000 amperes (11.3 MW) can be obtained without ripple.

The A.C. power system includes two 13.2 kV, 7 MW feeders from the Consolidated Edison Company, with special interlocks to prevent feedback from one to the other feeder in the event of a fault or ground. Interrupting capacity for the feeders is 250,000 kVA.

In the event of a trip fault on an A.C. breaker, all D.C. breakers are also tripped out, opening the feedback loop between the two feeders and preventing a D.C. motor overspeed.

Protection against synchronous motor field failure is provided by an inerlock that senses squirrel cage currents (that increase to a high level during a field failure) and trips the A.C. oil circuit breaker.

Four loops are available in the Facility for thermal-hydraulic testing. All can accommodate heated lengths up to 12.5 feet, or longer if necessary. All are serviced by the above power system. Two are "medium" to "high" pressure and two are "low" pressure loops. The "medium pressure" loop goes to 2400 psia and 550 gpm, and develops up to 600 feet of water head. The "ESADA" loop is rated to 3500 psia except for the present pumps, rated at 2500 psia and 650 F. Maximum flow is 400 gpm.

Two low pressure loops are also available. The Emergency Core Cooling System (ECCS) loop simulates a single PWR fuel assembly with corresponding representative reactor vessel volumes at the top and bottom, and permits testing of conditions for fuel rod cooling after a hypothetical loss-of-coolant accident.

The Low Pressure Thermal-Hydraulic Loop has been used for various fuel element heat transfer tests in the 100 psi range.

Each loop has its own instrumentation system and control room for loop operation; they all share the well-water system for heat rejection, the incipient burn-out detection recorders and a computer controlled data acquisition system. Each loop is equipped with pressure, temperature and flow recorders, controllers and indicators. The control rooms also contain the safety indicators and controllers for test section power, pumps, pump seal cooling and other auxiliary process parameters of the respective loops they service.

The Computer Controlled Data Acquisition System (CCDAS) is centered around a Hewlett Packard 2100 computer with a 32k word memory. The computer is interfaced with a HP7970B digital magnetic tape and a 15 MByte, HP7905 disc system for online and off-line data storage. A high speed, 112 channel, multiplexer data acquisition system with 8kHz sampling capability and a 200 channel low speed system with 10Hz sampling capability interface the analog experimental data with the computer. The multiplexer system is also capable of providing two analog signals under program control. These signals are used to activate safety relays in the test section power circuit. In addition there is a visual baragraphic display unit with a high signal relay operation capability. Any predetermined 80 signals can be connected to this unit. An ASR 33 teletype and a HP2640 CRT terminal are used for operator communication with the system. These units can be located in any of the loop operating areas. Other peripherals to the computer are a plotter, a terminal printer and paper tape punch and reader units. During experimental operation a core resident program enables data read out, recording and reduction.

Table 1.1

Heat Transfer Research Facility - Ratings of D.C. Power Supply

A. Ratings of Individual D.C. Generators:

	V_{Max}	Amps	Operating Time	MW
Old (2)	175	10,000	Continuous	1.75
	175	11,000	2 Hours	1.93
New (4)	230	6,000	Continuous	1.38
	230	7,500	2 Hours	1.73

B. Ratings of 2 Old Plus 2 New Generators Prior to Present Upgrading

Parallel Operation	175	32,000	Continuous	5.6
"	175	37,000	2 Hours	6.47
Series-Parallel Operation	230	22,000	Continuous	5.06
"	230	26,000	2 Hours	5.98

C. Ratings of 2 Old Plus 4 New Generators After the Current Upgrading

Parallel Operation	175	44,000	Continuous	7.7
"	175	52,000	2 Hours	9.18
Parallel With Rectifier Booster Operation	230	44,000	Continuous	10.12
"	230	50,000	2 Hours	11.5

Section 2

INTRODUCTION

2.1 GENERAL BACKGROUND

The safety and operational limits of a nuclear power reactor are strongly influenced by the steady state and transient thermal-hydraulic behavior of the heat generating core in the reactor. With the increased utilization of nuclear reactors for power generation, the need for extensive and reliable understanding of reactor thermal-hydraulics assumes a prominent role. In view of the apparent complexities, such studies are carried out by well controlled out-of-reactor simulation studies. The areas in which design and research information are needed for substantiating the operational and safety limits generally fall into the following categories:

a. Critical heat flux limitations under steady state and transient conditions over a wide range of process parameters anticipated under operating conditions and under postulated accident conditions of nuclear reactor cores.

b. Thermal-hydraulic behavior of large open lattice cores of Pressurized Water Reactors (PWRs).

c. Heat transfer behavior beyond critical heat flux under steady state and transient conditions.

d. Improved and detailed measurements of fluid local conditions within rod arrays.

e. Transient (under loss of coolant and loss of flow conditions) two-phase flow and heat transfer studies with improved instrumentation.

f. Heat transfer and design evaluations of emergency core cooling systems.

Most of these experiments are intended to provide design data as well as to substantiate and validate the theoretical models used in reactor analyses.

All the above studies, in general, are conducted by out-of reactor simulation. These thermal-hydraulic simulations are accomplished by creating reactor environments in closed loops. The heat generation in the simulated fuel rods is achieved by using electrical power. Thus the size of the model is usually dictated by the capacity of the power source available. Over the years, the Columbia University Heat Transfer Research Facility has been a major source of such thermal-hydraulic experimental data. There are four major thermal-hydraulic loops in use at this facility. All four loops receive power from the facility D.C. power system.

When the Columbia Heat Transfer Research Facility was established in 1950, two D.C. generators were installed with a rating of 3.5 MW. These generators were located in the basement, and they are identified as "basement generators". As time passed and the problems of modeling became more evident, the need for larger, higher-powered experiments became compelling. Accordingly, the Atomic Energy Commission, through the USAEC/AECL Cooperative Program for the Development of Heavy Water Reactors (with substantial contributions from the nuclear industry) funded an upgrading of the power system. Four identical reconditioned motor-generator sets, each with a short term rating of 1.725 MW, were purchased in 1968 and installed on the ground floor of the building. These sets are identified as "driveway generators". Two of the driveway sets were connected electrically to operate in parallel with the basement sets. The resulting available power then became 6.5 MW. Due to the high power densities in nuclear reactors, 6.5 MW enabled the simulation of only a fraction of a single fuel assembly of a typical PWR. However, the recently completed uprating of the power by commissioning all the driveway generators and upgrading the units in service brought the available power to 11.5 MW.

An analytical study funded by EPRI (RP-345, Task B) was performed to determine the experiment size required to satisfy modeling concerns, and to compare the equipment requirements for these experiments with the facilities available at Columbia, with particular emphasis on the nominal 11.5 MW power capability which would result from the uprating of the power supply. The results of this study indicated that valuable experimental work could be conducted with 11.5 MW, that would significantly reduce uncertainty due to modeling.

2.2 D.C. POWER SUPPLY UPRATING

This report presents the uprated power system designed and installed at the Heat Transfer Research Facility of Columbia University. The power previously available was generated from four D.C. generators which, when operated in parallel, generated 6.5 megawatts at 175 volts, the output power being continuously variable from zero to full power. The normal power range originally could be converted to operation at a maximum voltage of 230 volts D.C. by reconnecting two generators in series and operating in parallel with the remaining two generators; resulting in a maximum power of 6.0 MW.

The changes in the system are designed to generate 230 volts D.C., 50,000 amperes, a total generation of 11.5 megawatts. This output is accomplished by adding a voltage of 55 volts (by using rectifiers) in series with the two generators rated at 175 volts. In addition to the voltage boost in these two generators, two other generators are added to the system, which also will produce an output of 230 volts. The rectifiers will be operated in a three-phase, full-wave bridge mode at a voltage of 230 volts. This eliminates the usual side effects of current distribution from harmonics, together with the problem of irregular tube heating affected by eddy current and hysteresis losses, proximity effects and skin ratios which occur particularly in thick wall tubing. The six generators operating in parallel will generate a nearly pure D.C. voltage.

The increased power from the D.C. generators causes an increase in the power required from the 13.2 kV high voltage feeders, which are serviced by two separate 13.2 kV systems originating in the Hell Gate generating station of Consolidated

Edison Company. A finite amount of power is available on each of these two feeders; the maximum electrical load on any one feeder cannot be greater than 7 MW. The two feeders were originally tied, and only a single feeder could be used at any time. The circuit loading on the uprated system required a split in the two feeders to permit their simultaneous use.

The splitting of the two feeders causes additional problems. A feedback from one feeder to another could occur by a reversal of current in one D.C. generator. Should there be a fault on one feeder, the fault could have short circuit power obtained from the other feeder; the only impedance limiting the fault current would be the short circuit impedances of the generators, motors and the short circuit kVA available.

Protection for operation of the high voltage system must contain features for circuit interruption caused by a ground fault on any number of phases, short circuit on any number of phases, overload, undervoltage and reverse power. In addition, phase sequence on both feeders requires checking.

Starting requirements for the basement motor generator set containing generators numbers 1 and 2 are 760 amperes at 13.2 kV. The starting power is primarily reactive and this produces an apparent power of 17,400,000 volt-amperes. The initial starting current causes a voltage drop in the starting feeder of 9.5%. This excludes start-up of more than one motor at a time. All other motors are started by the use of the first motor generator set. The direct current power from one of its generators is connected to the generator of the set to be started and is rotated as a motor. The synchronous motor drive on the MG set is synchronized with the Consolidated Edison system and the driving generator is disconnected by means of a D.C. circuit breaker.

The D.C. system contains four motor-generators having a nominal rating of 1,500 kW, 250 volts, 6,000 amperes D.C. These generators have been assigned a short term rating of 20 minutes at 7,500 amperes, 230 volts, without overloading the synchronous motor drive or exceeding its pull-out torque. The four driveway generators are operated in parallel with the two basement generators connected in series with two rectifiers which boost the voltage of the combination to 230 volts. During operation, the rectifiers are maintained at their full voltage to minimize the harmonic voltages.

Protection for the operation of the direct current equipment is provided by D.C. circuit breakers on each generator and rectifier output. These breakers are operated by relays which activate the shunt trip circuit of the breaker control unit. These relays are activated by overload, short circuit reverse current, and by other instrumentation and sensors peculiar to an individual test.

In addition to the engineering phase required for the uprating, a cracked shaft on motor generator set number 2 in the driveway area was replaced by a new shaft. A new commutator was installed on the generator of the number 1 basement MG set, was properly aged and settled, and the generator was then re-aligned and re-balanced.

Section 6

EXPERIMENTAL FACILITIES SUPPORTED BY D.C. POWER SYSTEM

6.1 GENERAL

The Columbia D.C. power system is primarily intended to provide D.C. electrical power for carrying out thermal-hydraulic studies related to nuclear reactor technology. This section briefly describes some of the major experimental facilities currently in use at the Heat Transfer Research Facility of Columbia University. The major thermal-hydraulic loops that have been extensively used at the facility are

a. Medium Pressure Heat Transfer (MPHT) Loop
b. High Pressure Columbia-ESADA (CU-ESADA) Loop
c. ECCS Loop
d. Low Pressure Thermal-Hydraulic Loop

All the above loops are supported by a single Computer Controlled Data Acquisition System (CCDAS), and a well water system as a secondary heat sink. All the facilities are serviced by the D.C. power system. The Columbia-ESADA loop has an independent control room. The remaining loops share a single control area. However, there are independent heat exchangers and other process equipment for each of the loops.

6.2 MEDIUM PRESSURE HEAT TRANSFER LOOP

6.2.1 General

The loop is constructed of 300 Series stainless steel, with the main piping of nominal 3-and 4-inch diameters. The loop is shown schematically in Figure 6-1. The principal loop operational limits are given in Table 6-1.

Table 6-1

COLUMBIA MEDIUM PRESSURE HEAT TRANSFER LOOP OPERATIONAL LIMITS

Maximum pressure, psia	2400
Maximum test section flow, gpm	550
Head developed by pump, ft. of water	600
Test section housing length, inches	200
Maximum heated length of bundle, ft.	12.5
Test section housing ID, inches	6.5
Heat exchanger heat transfer area, $ft.^2$	250
Maximum secondary (well) water flow, gpm	600
Minimum well water temperature degrees F	64

The major components of the loop are a 100 hp Wilson-Snyder centrifugal pump, test section housing, mixing tee, heat exchangers, make-up system and purification system.

6.2.2 Primary Loop

The 100 hp Wilson-Snyder centrifugal pump provides the circulation of the coolant around the closed primary loop. The total flow leaving the main circulating pump splits with one part going through the test section and the remainder through the heat exchangers. The flow through the test section can be varied by means of a flow control valve which is manually operated from the loop control panel area. A second flow control valve in the heat exchanger line provides the test section inlet temperature control. The heat exchanger secondary flow, which also is controlled manually from the loop control area, provides additional control capability. The test section flow is measured by a venturi and a turbine flow meter prior to its entry into the test section housing. Within the test housing, the coolant removes heat from the heated test section and exits from the top of the test housing to unite in the mixing tee with the flow from the heat exchangers. The mixing tee provides a stable inlet temperature at the pump inlet and hence at the test section inlet. The heat exchanger branch of the primary loop consists of three heat exchangers and an air-actuated control valve. The heat exchangers have a total heat transfer area of 23.23 m^2 (250 square feet) with 15.42 m^2 (166 square feet) in the main heat exchanger. The loop is constructed in such a way that the heat exchangers can be operated singly or in any combination, thus providing a wide range of achievable subcooling even for low mass flow rates. The heat exchangers are tube-in-shell type, with the primary loop water on the tube side. The cooling water for the heat exchangers is obtained from wells on site. The secondary system is a once-through open loop.

6.2.3 Make-Up, Feed-Water and Pressure Control System

The make-up and feed-water system is used to fill the loop initially and to provide make-up water during operation to maintain loop pressure. Whenever the loop pressure decreases, a Bristol Model 1G500 FF-14 pressure recorder controller activates an air driven make-up pump (piston pump) to restore the pressure to the set value. Whenever the reference pressure exceeds the set control value, the overpressure is relieved by a "Mighty Mite" relief valve. The make-up system uses deionized water stored in a heated 93.3 C (200 F) intermediate tank which is kept filled automatically by city water fed through the main deionizing sys tem. In addition, there is a 13.8 gpm, John Bean Model M11 - 10 triplex piston pump (179 bar; 2600 psia) in the system to pressurize the loop relatively fast. This pump works only by operator control.

6.2.4 Purification System

The purification system is used to control the water chemistry in the primary loop. Part of the heat exchanger outlet flow is split into two streams. One stream is taken through the Graham Heliflow heat exchangers (Hx-2 and Hx-3 in Figure 6-1) and is used to cool the pump seal, finally merging with the pump inlet flow. The second stream is taken through two Parker Dual Coil heat exchangers (Hx-4 and Hx-5) and again split into two streams. One stream passes through the deionizers and joins the pump suction stream. The second stream is used to cool one of the test section chamber seals and is returned to the pump suction stream.

6.2.5 Well Water System

Water from wells on site is used as the heat sink in the loop heat exchangers and in the Parker Dual Coil heat exchangers (Hx-4 and Hx-5) and Graham Heliflow heat exchangers (Hx-2 and Hx-3) described in the purification system. The water from the wells is pumped by three centrifugal pumps, operated in parallel, to the inlet of the secondary side of the heat exchangers. The well water coming out of all the heat exchangers is discharged into the sewer.

6.2.6 D.C. Power System

Figure 6-2 shows a schematic of the D.C. power system for the Medium and High Pressure Heat Transfer Flow Loops in the laboratory.

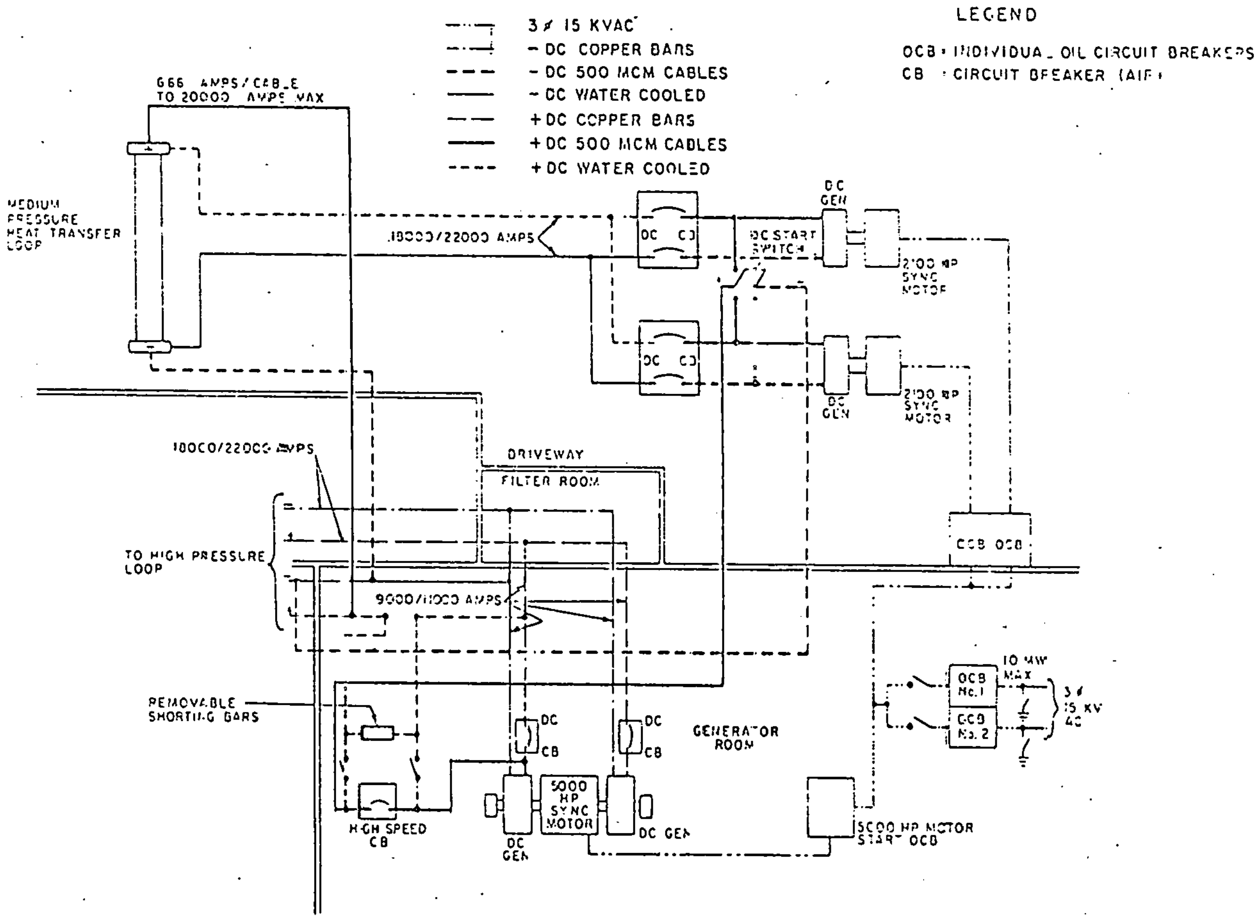


Figure 6-2

DC POWER SYSTEM

This system is described in detail in the earlier sections.

6.2.7 Flow Housing

The flow housing consists of four major components: the grid plate, the top adapter, the shroud box, and the bottom adapter.

The grid plate is machined from a nickel plate. It serves as a top electrical connection. The grid plate is accurately machined to maintain the tops of the rods in proper array geometry.

The top adapter locates the shroud box with reference to the heated rod geometry and provides a transition from the geometry of the rod bundle channel to an open arrangement for the coolant discharge.

The shroud box is made of four rectangular pieces of stainless steel, machined and fitted to hold the ceramic liner which actually forms the flow channel. The ceramic shroud liner is made of 98% dense Al_2O_3 in 508 mm (20") long sections, and ground to the desired dimensions within a tolerance of $\pm$ 0.0762 mm ($\pm$ 0.003"). Holes for pressure tap locations are drilled at selected points along the axial length of the flow channel.

The bottom adapter locates the inlet end of the flow channel with respect to the heated rods, and provides a region for coolant entry into the channel.

6.2.8 Test Sections

The heater rod diameter, length, rod array pitch, rod-to-wall spacing, number of rods in a bundle, and number of pressure taps and their locations, vary from test section to test section. The details covered in this section are of a general nature and are for a typical test section. Figure 6-3 shows a typical test section installed in the loop.

6.2.8.1 Heater Rods

The nuclear heating in fuel rods is simulated by passing direct

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electrical current through the rods. These rods are composed of top and bottom electrodes and a heated section. The heated section is fabricated from Inconel tubing or other suitable material. The tubing outer diameter is equal to the OD of fuel rods being simulated. The tubing wall thicknesses vary over the range 0.254 mm (0.010") to 1.27 mm (0.050") to achieve the desired radial and axial heat flux in the bundle and to achieve a proper overall test section resistance to be compatible with the power system. In all cases resistance measurements of each finished rod are made using a Kelvin bridge, and the individual heater rod resistances are used to compute the true radial power distribution.

The heated rods are filled with ceramic support cylinders. These cylinders are made from 98% dense Al_2O_3 ceramic, 76.2 mm (3") long with the OD ground to match the ID of the tube. The inside of the heated tube is at atmospheric pressure during testing. A 6.35 mm (1/4") hole through the bottom electrode and the ceramic cylinders permits access for instrumentation. The ceramic cylinders prevent collapsing of the thin-walled tubing due to external pressure.

The top electrode is made of a solid piece of nickel rod with the same OD as the heated section. The top end of the nickel rod is ground with a self-locking taper which fits a matching conical hole in the grid plate.

The bottom electrode is made of nickel tubing over the length where the rod is in contact with high pressure loop water. The bottom electrode is joined to copper tubing external to the pressure seal. A typical rod assembly is shown in Figure 6-4.

6.3 HIGH PRESSURE COLUMBIA-ESADA LOOP

6.3.1 Facility Description

The test facilities consist of a main circulating loop, a side stream purification system, auxiliary cooling system, and a control system. A layout of the loop is shown in Figure 6-5. A flow

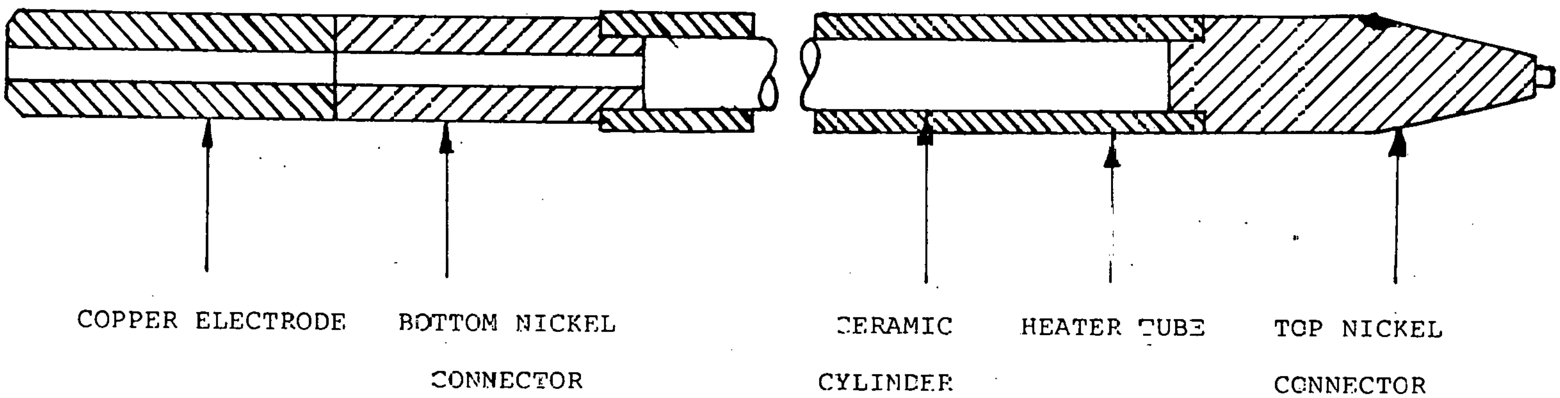

Figure 6-4

TYPICAL ROD ASSEMBLY

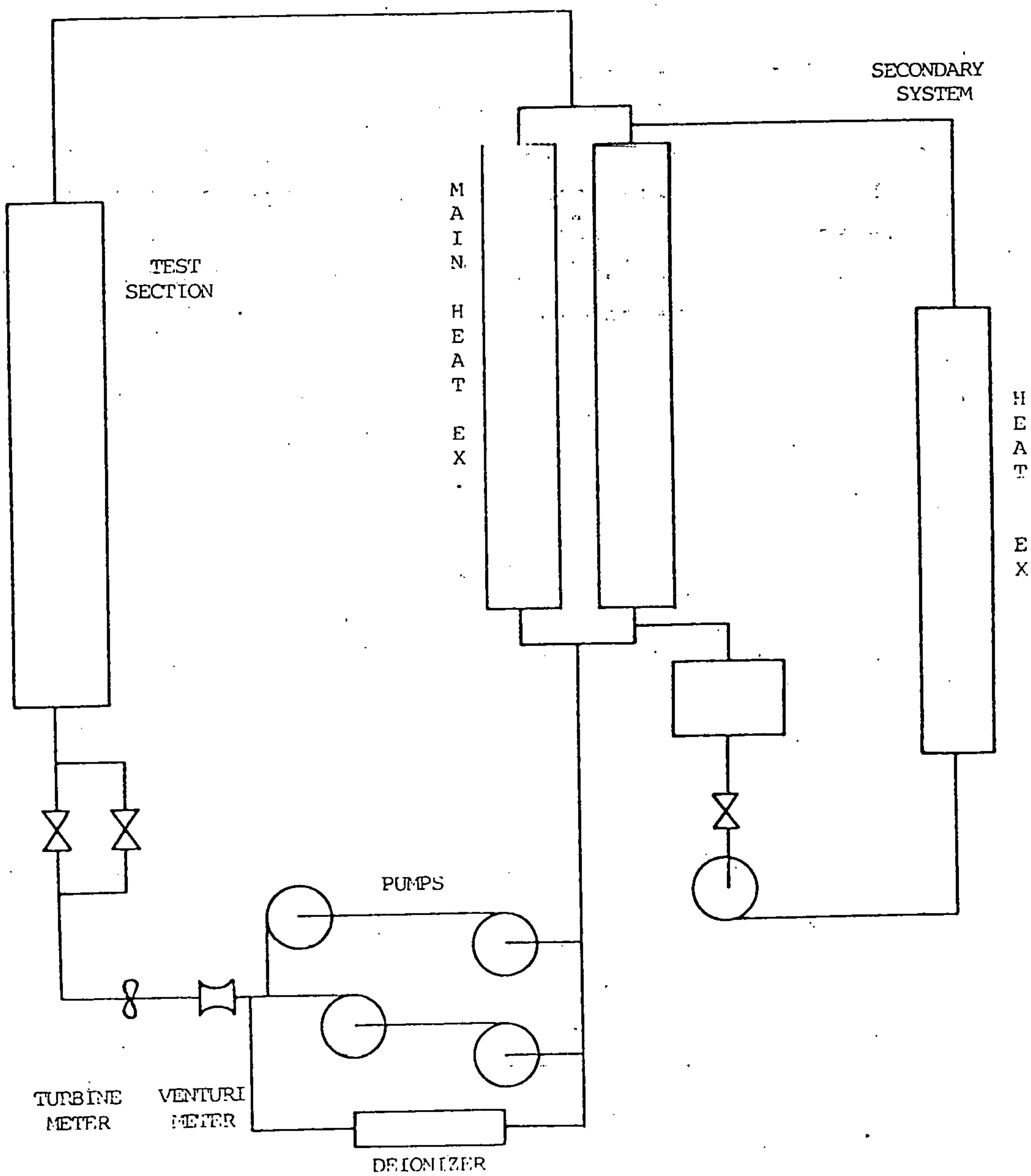

Figure 6-5

COLUMBIA - ESADA LOOP

diagram of the main circulating loop and auxiliary systems is illustrated schematically in Figure 6-6. The main circulating loop components are the test section housing, main heat exchanger, circulating pumps, and interconnecting piping. These loop components were designed according to the ASME Boiler and Pressure Code and the following specifications:

• Loop flow rate, maximum	400 gpm
• Loop operating pressure*, maximum	3500 psia
• Loop operating temperature, maximum	700 F
• Main loop cooler heat removal capacity	5 MW

6.3.2 Test Section

• Inlet temperature, maximum	650 F
• Outlet temperature maximum	700 F
• Rod bundle length	8 or 14 ft.
• Housing length, maximum	16 ft.

The test section consists of a pressure housing, ceramic flow channel, flow channel shroud, "O"-ring pressure plugs and electrical power terminals.

The pressure housing is made up in two sections. The lower section, 10 ft. long, is employed when 8 ft. long rod bundles are tested. The upper section, 6 ft. long, is added to the lower section when the 14 ft. long rod bundles are used. Each section was constructed from a single forging with integral flanges equivalent to type 316 stainless steel 8-inch schedule 160 pipe and welded neck flanges rated for 3500 psia service.

The ceramic flow channel is made from precision machined ceramic (99% Al_2O_3) sections, each about 12 inches long, placed inside the flow channel shroud. The ceramic prevents galvanic corrosion between the heater rods and flow channel shroud. The ceramic channel extends about a foot beyond the rod bundle heated

* Refers to all components except the pumps which were designed for 650F inlet temperature and 2500 psia system pressure.

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length both upstream and downstream to provide a constant inlet and outlet geometry and prevent adverse entrance and exit flow effects.

The flow channel shroud is constructed from type 17-4 PH stainless steel plates bolted together to form a rigid square housing to support the ceramic flow channel sections. The 17-4 PH stainless steel material was chosen to more closely match the expansion rate of the ceramic flow channel, thereby eliminating potential bypass flow. The flow channel has two transition adapters, one at each end. Both adapters offer a transition between circular and square channel geometries. The bottom adapter wall has eight 1-inch diameter holes equally spaced circumferentially to evenly distribute the incoming flow. The top adapter is bolted to the rod bundle top grid plate and enlarges the flow channel to allow steam-water mixture to flow through the grid plate perforations. Water coming from the test section housing inlet pipe flows down between the annulus formed by the shroud and the pressure housing wall, then passes through the shroud bottom adapter holes and turns upwards inside the square flow channel containing the rod bundle test section. Continuing upward, it flows through the grid plate and into the pipe connecting the test section and main heat exchanger.

Seals for the heater rod penetrations at the bottom are provided by plugs containing Viton "O"-rings, which are spaced to form two chambers which provide cooling to the "O"-rings. These chambers step down the pressure differential between the test section internal pressure and atmosphere. The top chamber is supplied with external cooling water (well water) at 500 to 900 psig.

6.3.3 Main Heat Exchanger

Two parallel shell and tube heat exchangers, operated as evaporative coolers, serve to remove the heat input from test section and to maintain a constant test section inlet temperature. The heat removed from the primary coolant, which flows inside the tubes, is transferred to the low pressure boiling water in the shell side of the heat exchanger. The steam generated in the shell side is piped to another heat exchanger where it is condensed to a subcooled temperature and recirculated back to the main heat exchanger.

The main coolers are capable of removing at least 5 MW (17×10^6 Btu/hr) of heat by vaporizing about 14,000 lbm/hr (32 gpm) of water on the secondary side. This amount of heat is removed when the primary side is operating between 400 F and 650 F and the secondary at saturation conditions of 150 psia. The rate of heat removal is controlled by manually adjusting the flow to the secondary side of the heat exchanger.

6.3.4 Main Circulating Pumps

To meet the operating requirements of the loop and permit flexibility of operation, four canned motor pumps are arranged in two sets connected in parallel. Each set has two pumps connected in series in order to provide a maximum developed pressure head of 580 feet of water. The pumps, Westinghouse Model A-150-D1, are constructed of Type 316 stainless steel. Each pump is rated at 150 gpm against a head of 300 feet, for 2500 psi and 650 F service.

6.3.5 Interconnecting Loop Piping

The primary loop piping is constructed of seamless 4-inch schedule 160 Type 316 stainless steel pipe. The piping is rated at 3500 psi and 700 F. Because of the large range of operating flows (40-400 gpm), two different size valves connected in parallel are used to control flow. These valves are 4-inch and 1-inch Y-type valves located upstream of the test section inlet, and are operated manually to adjust loop flows as required. In addition, a Venturi type BIF flow nozzle and a Potter type turbine meter are installed in the loop piping to measure flow.

6.3.6 Side Stream Purification System

A portion of the main loop circulating water flows through a side stream cooler and two parallel purification columns containing a mixed bed of ion-exchange resin. The system is capable of purifying loop primary water at a rate of 3 loop volumes per hour in order to maintain low chlorides (0.10 ppm), maximum water resistivity of 0.5 x 10^6 ohm-cm, and low solids concentrations. It is also equipped with an in-line conductivity cell, and a sampling cylinder. This cylinder is employed to obtain loop water samples for chemical analysis and to inject the necessary chemicals to adjust loop water chemistry.

6.3.7 Auxiliary Cooling System

The main heat exchanger secondary side and the main circulating pumps require a closed loop recirculating cooling system in order to maintain low chloride concentrations to avoid stress cracking corrosion of the heat exchangers surfaces. Each system consists of a circulating pump and intermediate heat exchanger. Cooling water to the secondary side is provided from a well in the facility or the city water supply lines.

6.3.8 The Control Room

The control room contains loop controls and instrumentation needed to perform the following functions:

1. Control loop temperature at any desired value between 100 F and 650 F.
2. Control loop pressure at any desired value between 100 psia and 2500 psia. The control is applied to a totally liquid filled system.
3. Control loop flow between 40 and 4000 gpm.
4. Provide temperature protection for pumps in the form of visual and audible alarms.
5. Provide over-pressure protection for the system in the form of relief devices and audio-visual alarms.
6. Monitor water supply tanks liquid levels with audio-visual high and low level alarms.

In addition, the DNB detection recorders are located in this control room. Loop instrumentation is shown schematically in Figure 6-6. Temperatures are monitored by thermocouples in conjunction with single-point circular chart and multi-point strip chart recorders. Test section inlet and outlet temperatures are measured with thermocouples and recorded by the CCDAS as described in Section 6.7.

The loop pressure is remotely controlled by an air operated make-up pump and a letdown backpressure regulator. Test section inlet and outlet pressures are measured with precision Bourdon tube gages. The outlet pressure is also recorded in a circular chart pressure recorder. This recorder is provided with adjustable high and low limits which activate a letdown solenoid valve and the make-up pump, respectively. A gas pressure controlled letdown valve is normally used to steady pressure control during minor pressure changes. In addition, the loop pressure can be controlled manually with a dump valve.

Pressure drops across the test section heated length are measured with Meriam high pressure, differential pressure transducers.

Test section inlet flows are adjusted manually with a combination of a 4-inch and 1-inch valves connected in parallel. Flow measurements are made by a Venturi type flow meter, and a Potter turbine meter. The Venturi flow meter differential pressures are measured with two high pressure Merian differential type manometers. High flows are monitored with a mercury manometer, and low flows with an oil manometer with oil having a specific gravity of 2.95. The turbine meter has a digital read out in cycles per second which is proportional to flow rates.

Test section power is recorded with the data acquisition system. Volts are measured across voltage taps built in a heater rod, or across the test section power terminals. Currents are monitored by 10,000 A, shunts located in each of the M-G sets main electrical bus.

6.4 GENERAL INSTRUMENTATION FOR MPHT AND CU-ESADA LOOPS

The instrumentation for the MPHT and CU-ESADA loops is similar. This section describes the general instrumentation in use.

However, the exact test instrumentation depends on the scope and purpose of individual tests. Typical measurements include heater rod temperatures, test section power, coolant flow rate and coolant temperatures. Test section pressure drop and subchannel temperatures can also be measured. Transient measurements are also carried out as required by tests. Table 6-2 gives a general tabulation for typical tests. The instrument signals are processed by the CCDAS as described in Section 6.7.

6.5 EMERGENCY CORE COOLING STUDIES (ECCS) LOOP

This loop was primarily built to study the top spray-emergency core cooling as designed for Indian Point-1 reactor. However, it is flexible enough to be used for alternate emergency core cooling system studies. The test loop was designed to simulate a single reactor fuel bundle assembly with representative reactor vessel volumes. The main components of the loop consist of:

a. Test Section Assembly
b. Hot and Cold Leg Piping
c. Core Bypass Piping
d. Upper and Lower Volume Tanks
e. Steam System
f. Core Cooling System
g. Steam Separator and Condenser
h. Instrumentation
i. Data Acquisition System

A flow schematic showing the test loop configuration for the cold leg breaks is shown in Figure 6-7.

6.5.1 Test Section Assembly

The test section assembly is comprised of an upper and lower pressure housing, with the latter containing components simulating a quarter section of the upper control rod assembly and the former containing the simulated fuel bundle assembly. Surrounding the outer housing three sets of strip heaters are installed to provide for the possibility of more accurate modeling of the canister to neighboring canister boundary condition.

Table 6-2

INSTRUMENTATION DESCRIPTION

Measurement	Type	Manufacturer and Range
Flow	Venturi	BIF Corp. 40-450 gpm. Differential pressure read on a 60-inch Meriam Manometer
	Turbine Meter	Flow Technology Inc. 20-650 gpm.
Absolute Pressure	Gauge	Heise Bourdon Tube Co. 0-2600 psi, 2 psi div.
	Strain Gauge	BLH Inc.; 0=5000 spi
	Bourdon Tube	Texas Instrument Co. 0-5000 psi
Differential Pressure	Strain Gauge	BLH Inc.; 0-10 psi, 0-20 psi
	Manometer	Meriam Instrument Co. 60-inch Manometer with 0.1 inch division
Test Section Inlet and Outlet Temperature	RTD	Weed Inc.; 400-650 F
	Thermocouple	Conax Corp. Iron/Constantan; 0-1400 F
Rod Temperature	Thermocouple	Conax Corp. and C.S. Gordon Chromel/Alumel; 0-2000 F
Voltage	Voltmeter	Hewlett Packard Co. 0-240 V
Current	Shunt	Westinghouse Corp. 0-6000 Amp

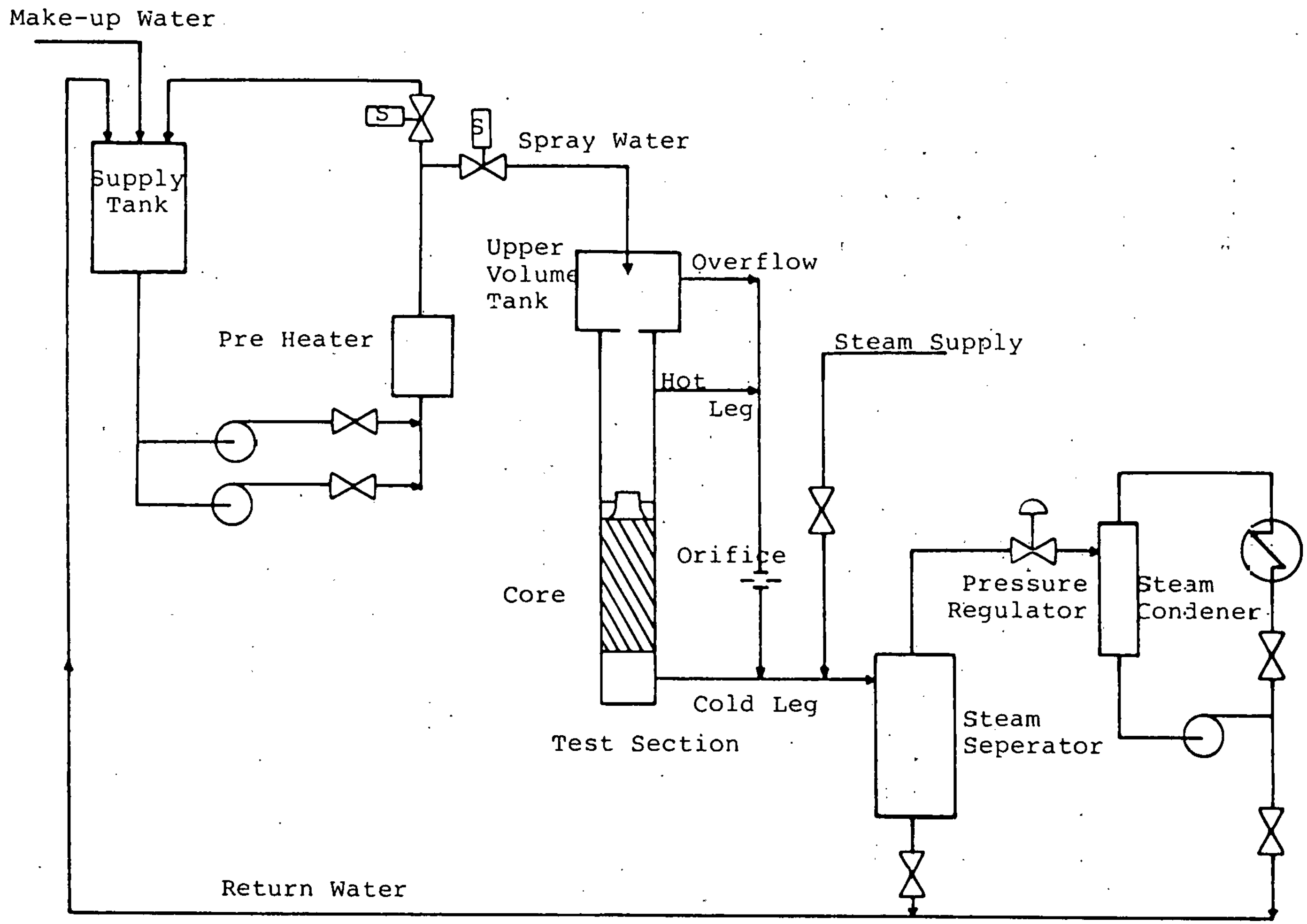


Figure 6-7

ECCS Loop Schematic

6.5.2 Upper Housing

The upper housing is a square cross-sectioned pressure-retaining assembly with interior dimensions of 6.238 in x 6.238 in x 113.45 in long. The interior space contains two stainless steel slabs, each representing half the thickness of two control rod blades a quarter round tank representing the control rod guide tube, the reactor upper grid plate and a fuel element top nozzle.

6.5.3 Lower Housing

The lower housing is a square-cross-sectioned pressure-retaining assembly with an interior configuration following the outline of a fuel element assembly. The interior dimensions were selected to represent the perimeter envelope, including clearances, of a single fuel assembly.

Located within the housing is a production fuel assembly canister and top nozzle assembly, with 173 heater rods simulating a complete fuel bundle. Spacer pads on the outside of the shroud keep the heater assembly centered within the housing, while maintaining the proper water gap clearance.

A plate between the upper and lower housing simulates the fuel element upper grid plate and provides a common electrical ground for the heat rods. Attached to the bottom end of the lower housing is the bottom extension assembly. This assembly contains the simulated reactor cold leg piping and the sealing glands for the heator rods.

6.5.4 Core Bypass Piping

The reactor design incorporates a core bypass flow area from the lower to the upper plenums and to simulate this region the test loop includes a small orifice sized to provide the equivalent pressure drop characteristics.

6.5.5 Upper and Lower Volume Tanks

The test loop simulates the reactor vessel upper and lower plenums by the use of tanks with volumes sized on a single fuel assembly basis.

The upper volume tank connects directly to the top of the upper housing. The bottom of the tank simulates the reactor upper diaphragm plate and contains two 7/8-inch diameter holes to control the flow of core cooling water into the test bundle. The top of the tank contains the spray nozzles for injection of core cooling water into the test section.

The lower volume tank is separated from the test section housing, due to space limitations at the testing facility. In the cold leg break configuration the lower volume tank is not needed and is removed from the system.

6.5.6 Steam System

To simulate the containment conditions immediately after the blowdown phase of a LOCA, an auxiliary steam supply system is connected to the test loop. Actual steam flow into the loop is controlled by means of a manual throttling valve.

The pressure in the loop is maintained by an automatic regulating valve located between the steam separator and condensing system. To improve the reliability of the steam flow measurements, the steam separator is used to minimize the amount of water carry-over into the turbine meter.

6.5.7 Core Cooling System

To provide a constant flow of core cooling water, a separate water flow loop with positive displacement pumps is used. A preheater is used to raise and hold the water at the desired temperature. Electric solenoid valves are used to switch the flow of water from the recirculation and preheat mode to the test section spray mode. In the latter configuration water is pumped from the supply tank, through the preheater and into the upper volume tank spray nozzle.

6.5.8 Instrumentation

The instrumentation used on the loop consists of thermocouples, differential pressure transducers, pressure transducers and turbine flow meters. The output signals (voltages) from all of the significant parameters were recorded in the data acquisition system. A listing and description of the significant parameters measured is given in Table 6-3.

6.6 LOW PRESSURE THERMAL HYDRAULIC LOOP

6.6.1 General

This loop was constructed in the 1950's to conduct thermal-hydraulic experiments for the duPont production reactors at Savannah River. It has been used over the years for different customers in a variety of configurations. The power supply and computer-controlled data acquisition system, described in this report, also service this loop. The principal features of the loop are described below, and a schematic is shown in Figure 6-8. The principal features are as follows.

6.6.2 Test Section Station

Since the loop is low pressure, there is no need for a test section pressure housing. The flow channel also serves the pressure housing. Test section heated length, therefore, is not limited by an existing pressure housing. Test sections with a heated length of 12 feet have been run, and there is headroom available to accommodate much longer lengths.

6.6.3 Pumps

There are three pumps in the loop, as listed below. These are usually operated in parallel, however they may be run in various series or series parallel arrangements as required.

Pump Number	Piping Size	gpm	Head (ft)	Motor H.P.
1	3 x 3	250	200	20
2	3 x 3	250	200	20
3	3 x 2.5	250	275	40

Table 6-3

INSTRUMENTATION INPUT INTO HIGH SPEED DATA ACQUISITION SYSTEM ECCS LOOP

Parameter	Sensing Element	Location
Flow	Turbine Flow Meter	1. Primary Spray Water
		2. Steam Separator Return Water
		3. Steam Condenser Condensate
		4. Steam Condenser Spray Water
		5. Steam Separator Steam Output
Pressure	Pressure Transducer	1. Top Volume Tank
		2. Test Section Housing
		3. Steam Separator
		4. Steam Separator Output Steam
		5. Cold Leg Piping
Temperature	Iron/Constantan Thermocouple	1. Primary Spray Water
		2. Test Section Housing
		3. Steam Separator
		4. Heat Exchanger Primary Side Inlet
		5. Heat Exchanger Primary Side Outlet
		6. Steam Separator Output Steam
	Chromel/Alumel Thermocouple	1.–5. Canister
		6.–82. Heater Rods
Power	Calibrated Resister (Voltage)	1.–4. Voltage Drop Across Ballast Resisters For Rod Groups 1, 2, 3, & 4
		5. Voltage Drop Across Group 5 Rods
	Shunt Resister (Amperage)	6. Voltage Drop Across Preheater
		7. Total Heater Rod Current
		8. Preheater Current

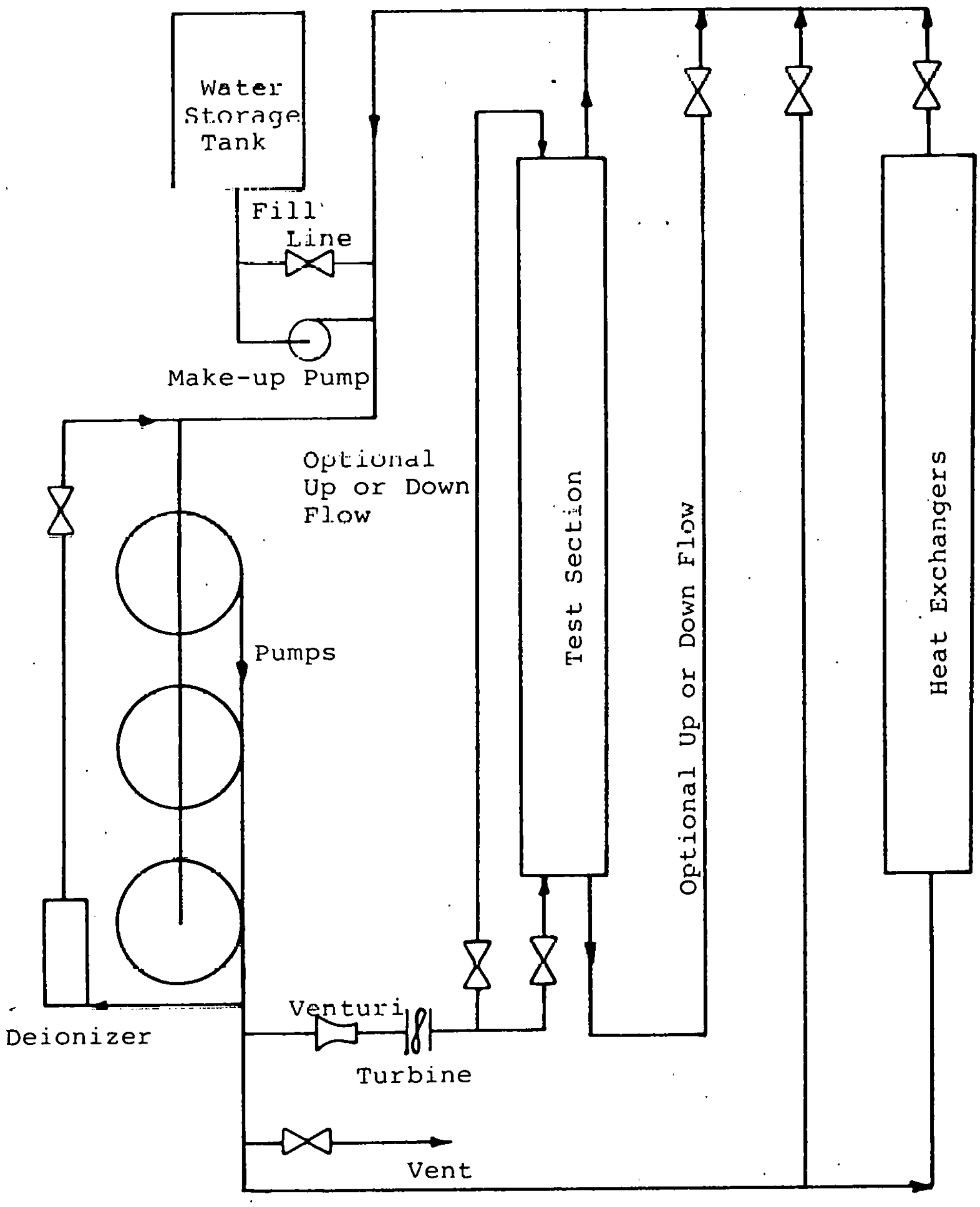


Figure 6-8

Schematic of Low Pressure Loop

6.6.4 Heat Exchangers

There are two shell and tube heat exchangers. The total available capacity is 153,515 Btu/hr F. Well water is used on the shell side. It is available at 64 F and 450 gpm.

6.6.5 Flow Control and Measurement

The flow is measured using the venturi and turbine meters from the medium pressure loop. Adaptors are used to accommodate the high pressure piping used for the meters to the low pressure loop piping. The flow is controlled manually by adjusting a flow control valve.

6.6.6 General Loop Information

The loop is filled with deionized deaerated water from a heated storage tank. During operation, the water quality is maintained by using a side stream deionizer. The loop is filled by gravity. Pressure is maintained by a feed and bleed system. A make-up pump adds water when the pressure is low and water is vented when the pressure is high. The primary piping and other components are fabricated from stainless steel.

6.7 COMPUTER CONTROLLED DATA ACQUISITION SYSTEM (CCDAS)

The CCDAS is comprised of the following:

a. Hewlett-Packard, HP2100A computer with 32K core, 2-channel direct memory access, time base generator and floating point hardware.

b. HP7970B digital magnetic tape unit. 9 track, 800 bpi, 37.5 i.p.s. speed.

c. HP7905 15MByte disc subsystem

d. HP2313B subsystem with a programmable pacer dual channel digital-to-analog converter and multiplexers for 112 channel differential input for high speed data acquisition system. (8000 samples per second for sequential channel scanning, 45 kHz single channel sampling).

e. HP2322 subsystem. Low speed data acquisition system with crossbar scanner capable of scanning 200 differential inputs at the rate of 10 samples per second.

f. HP2401 digital voltmeter

g. HP2895B paper tape punch unit

h. HP2748B paper tape reader

i. HP2640 CRT display terminal

j. HP2762A terminal printer, 118 columns, 30 characters per second.

k. ASR 33 teletype

The computer can operate under either Real Time (HP-RTE-2) or a Magnetic Tape System (MTS). Experimental data is recorded on magnetic tape. Figure 6-9 shows the CCDAS schematically.

The software consists of a main program which controls the use of a number of acquisition and data reduction subroutines. The main program is in on-line contact with the operator through the ASR 33 teletype which is located in the control room at the facility where the test is being conducted. According to the program option selected, the computer will initiate single sequential scan, multiple sequential scan, scan a random block of channels and perform a pre or post-test reduction of certain variables. The data reduction option picks the appropriate scan from the magnetic tape, reduces the data to engineering units, makes comparisons of temperatures (from thermocouple and RTD) and flows (from turbine flow meter and Venturi meter readings), calculates heat balances, pressure drop and test section power and heat flux. The output is returned on the teletype and the terminal printer.

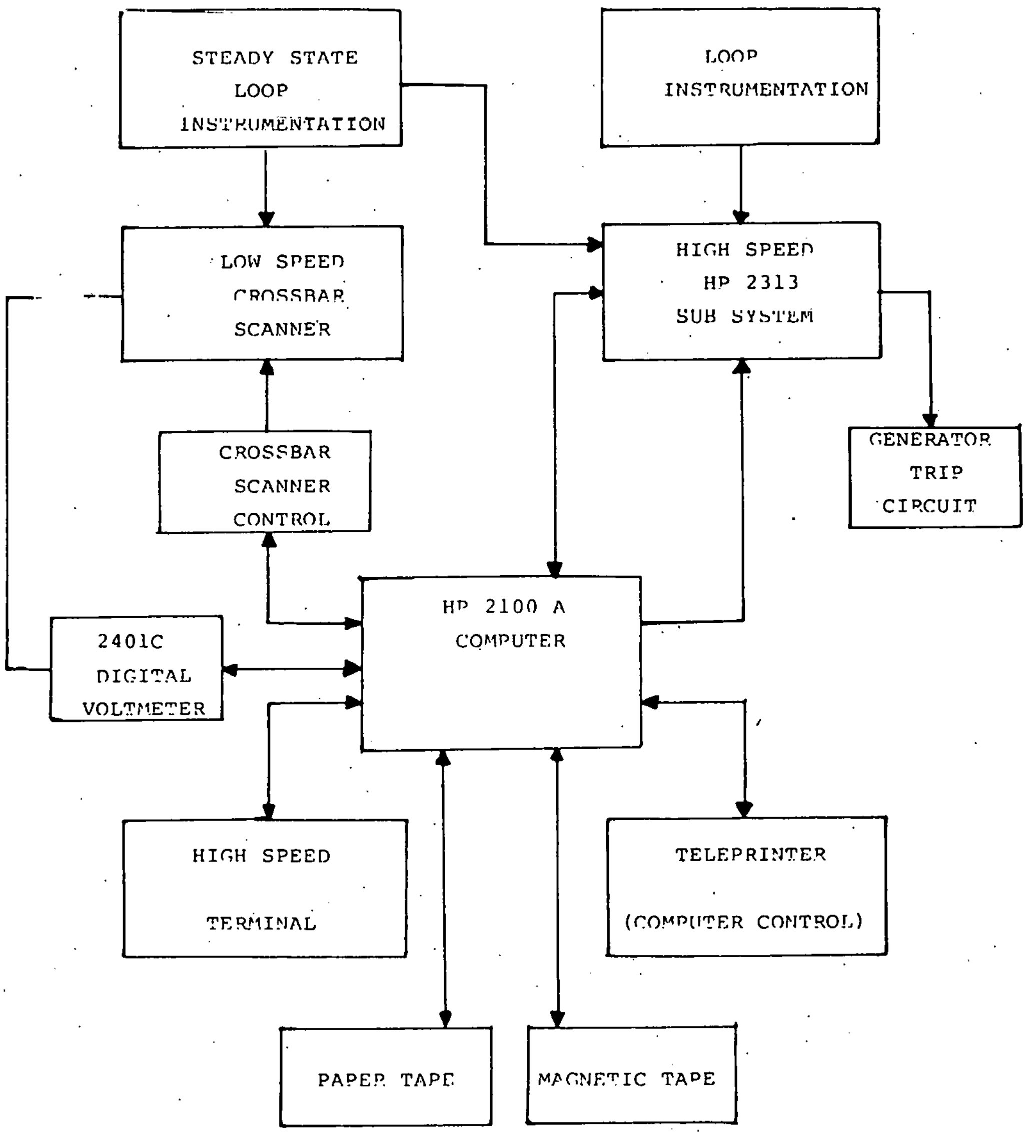


Figure 6-9

COMPUTER CONTROLLED DATA ACQUISITION SYSTEM SCHEMATIC

APPENDIX--I

APPENDIX I

McDONNELL DOUGLAS ASTRONAUTICS
COMPANY RESPONSE

MCDONNELL DOUGLAS ASTRONAUTICS COMPANY
ST. LOUIS DIVISION
Box 516, Saint Louis, Missouri 63166

28 September 1981

Dr. P. Y. Hsu
TPE-II Program Manager
EG&G
P.O. Box 1625
Idaho, Falls, Idaho 83415

Subject: Thermal Hydraulic Thermal Mechanical Testing Facilities

1. Enclosed please find our response to your questionnaire of 11 Sept 1981. Our facilities are primarily capable of providing surface heat fluxes to components. The major facilities are Quartz/Tungsten lamps, Graphite heaters, and a 15 kw CO_2 cw laser. These facilities are listed in the attached table in a format that responds to Question 1 of your letter.

2. The cost of operating the facilities is included in our overhead rates and only the manpower required for operation is charged to the user. Our composite labor rate is approximately $50/hour. The number of personnel required to staff a test is dependent on the exact nature and duration of the tests. Long duration tests are generally automated to reduce staffing requirements.

3. In regard to your Questions 2 and 3, we do not have high pressure coolant loops, since most of our facilities have been used for heating devices where low-pressure coolant is used.

4. I hope the information we have provided is useful to you and I would be glad to discuss potential applications of these facilities to specific tests.

Sincerely,

MCDONNELL DOUGLAS ASTRONAUTICS COMPANY

Clarence Trachsel

Clarence Trachsel 314-232-4867

Attachment

THERMAL-HYDRAULIC THERMOMECHANICAL TESTING FACILITIES

Heat Source	Quartz/Tungsten Lamp	Graphite Htr. Elements	15KW, CO_2 Laser
Test Environment (Vacuum, Inert gas)	Vacuum, Inert Gas, Air	Vacuum, Inert Gas	Vacuum, Inert Gas, Air
Test Specimen Size Accomodated	As Req'd △1	As Req'd △2	Up to 1x1 Meter
Materials Accommodated	Non Toxic, Not Radioactive	Non Toxic, Not Radioactive	Non Toxic, Not Radioactive
Temperature/Power Range	Element Temp. to 3000°C Maximum Power: $\frac{50\ \text{WATT}}{\text{cm}^2}$	Element Temp. to 2400°C Maximum Power: $\frac{300\ \text{WATT}}{\text{cm}^2}$	12.5KW Max. Output Pwr. 10.0KW/cm^2 Max. Flux
Rise/Decay Times	200 ms.	200 ms.	Approx. 100 ms.
Capabilities for Active Cooling	Low Pressure H_2O and LN_2	Low Pressure H_2O and LN_2	Low Pressure H_2O and LN_2
Facilities Instrumentation	△3	△3	△3
Facility Availability	100%	100%	20%
Support Facilities	△4	△4	△4

△1 Vacuum chambers from 2½' dia. x 5½' long to 30' dia x 35' long.

△2 Existing elements are 26 x 26 inches and 5 x 72 inches. Can be set up in any of above chambers.

△3 Automated data acquisition system. Parameters as required.

△4 Complete chemistry, metallurgy, leak detection, laboratores and machine shops.

APPENDIX J

LAWRENCE LIVERMORE NATIONAL LABORATORY RESPONSE

MFTF
MIRROR FUSION TEST FACILITY

September 28, 1981
MF-7.0

P. Y. Hsu
TPE-II Program Manager
Fusion Technology Program
P. O. Box 1625
Idaho Falls, Idaho 83415

Dear Mr. Hsu:

Reference: THERMAL-HYDRAULIC THERMOMECHANICAL TESTING FACILITIES

Enclosed is our brief reply to your request for information on the existing test facilities that might have some usefulness to your program.

Please do not hesitate to call if additional information is desired.

Sincerely,

Victor N. Karpenko
MFTF Project Manager

VNK/lo/
0250p

Enclosures

1. A. The heat source for the HVTS Facility is a neutral source whose beam contains neutrals and ions of deutrium.

B. The test environment is a vacuum with 10^{-7} torr capability with H_2 or D_2 gas.

C. The diameter of the target chamber is 61 inches. Without disassembling the target tank, access to the chamber is possible through a 20 inch diameter port of the diagnostic chamber. The 55 inch diameter diagnostic tank may also be used to fixture test pieces. The testing area is limited by the size of the beam in the target chamber area. The beam (LBL source) is elliptical (96 cm X 28 cm at the .5kW/cm^2 constant power density line.)

D. Any non-toxic metal or ceramic is acceptable as a test specimen.

E. A typical beam profile is enclosed. The maximum expected flux available is 9 kW/cm^2 with a maximum power of 4.6 megawatts.

F. The rise time for the neutral beam full power is .2 - .3ms, the decay time is .02 - .03ms.

G. The beam dump now in place (Feb. 1982) has a water flow system capable of 750 psia at 700 gpm for a 5 minute duty cycle with the beam on for 30 seconds.
If the test sample occludes part of dump, water could be rerouted from the shaded portion of the dump to the test piece (with some plumbing modifications). Also available is 300 gpm of unfiltered LCW at an inlet pressure of 75 psi.

H. The 75 psi LCW system is instrumented for calorimetry. The high pressure system is partially instrumented for calorimetry. Assorted vacuum transducers are placed at various places on the vessel.

I. There is at this time an informal schedule for the use of the HVTS facilities. The schedule would allow testing of outside samples only in conjunction with source testing.

J. A machine shop is adjacent to the HVTS. LLNL has some of the most advanced metallagraphic and optical facilities in the United States. Across the street from the HVTS is the headquarters for the Magnetic Fusion Energy Computer Center (MFECC) which has at the labs disposal, 2 CRAYls and a CDC 7600.

K. The HVTS has a staff of 7 technicians and a half time physicist and a half time engineer. At anytime, the HVTS may borrow an engineer or technician temporarily from another project if their skills are needed.

2. A. The facility has no blowdown capability. A diagram of the HVTS is enclosed. The circulation loop cannot handle two phase flow at the water outlet of the HVTS vessel.

   B. LCW is the primary coolant although a LCW-glycol mixture could be added since the cycle is closed loop.

   C. The neutral sources soon to be available are the 4.6 MW ORNL prototype 30 second source and the 3.6 MW LBL prototype 30 second source. Also available will be the RCA 5.4 MW - 1/2 second source.

   D. The maximum flow rate is 700 gpm at 750 psi, with full pressure drop to 40 psi allowed. The maximum water is available only if the test sample intercepts the entire beam.

3. A. Almost no neutron flux is generated by the neutral beams interaction with vessel materials.

   B. The total length of the tanks is approximately, 30 ft. with a minimum diameter of 55 inches.

   C. See response 3a.

   D. Almost no gamma heating takes place in the vessel.

   E. The facilities will be used to test and condition the sources for MFTF-B.

   F. The chamber is at room temperature and 10^{-6} torr.

   G. The Lab has extensive machine shops, welding shops and vacuum labs. Many computer terminals are available which are linked to MFECC.

   H. There are 4 electronic technicians, 2 physics technicians and a technician supervisor supporting the HVTS.

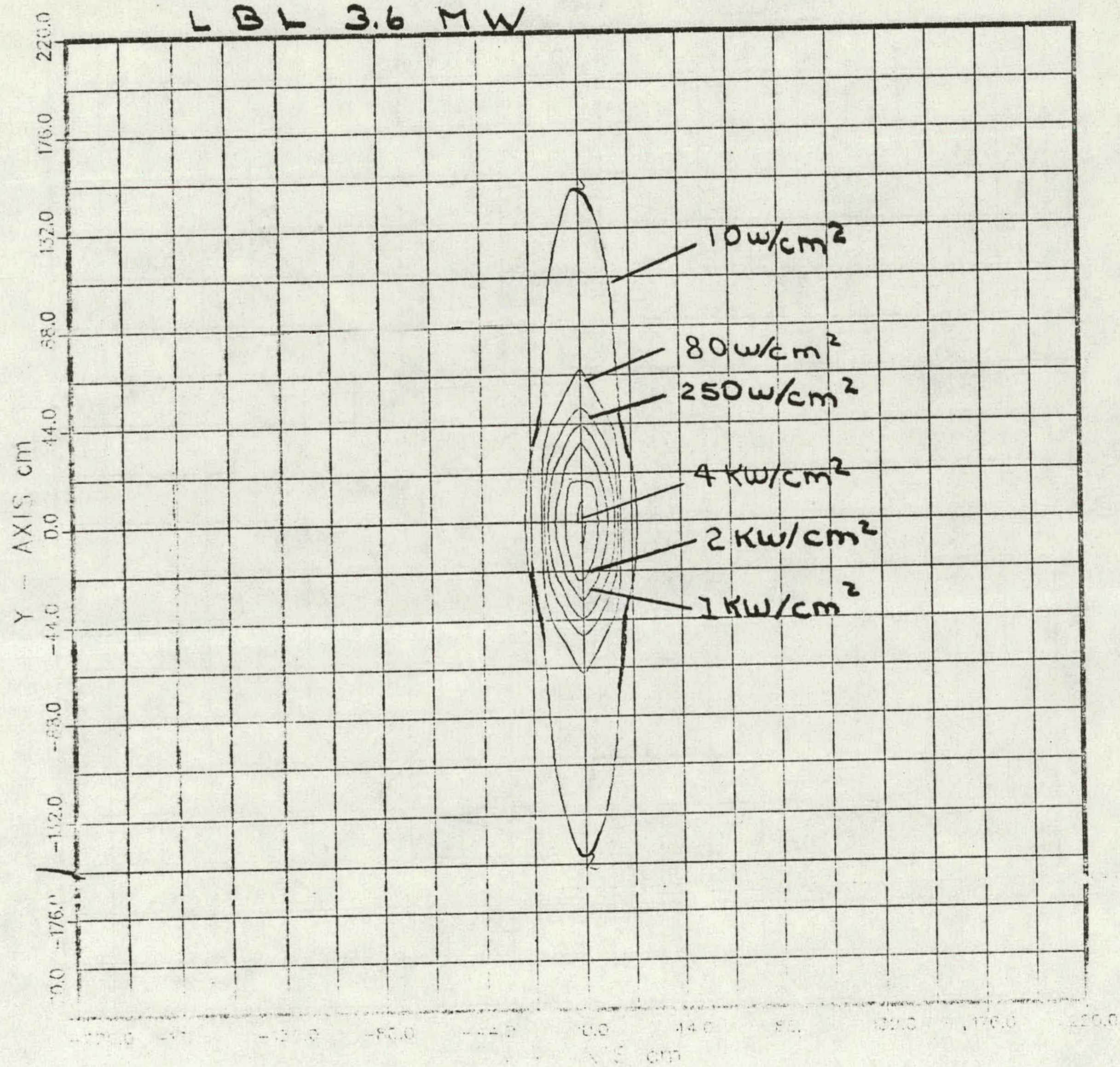
AG/rlh
0160h

POWER DENSITY CONTOUR PLOT

LBL 10*40 .4 1.4-90 .5-5.0-10 FXY-13 FXY-INF

SOURCE POWER-kW 3240.00 360.00

LBL 3.6 MW



BEAMLINE NO. 1	PLOT ORIGIN	0.	0.	1150.
TWO DIM. PLOT NO. 1	PLOT NORMAL	0.000	0.000	1.000
	MAX. POWER DEN.	4.660		

CONTOUR LEVELS	0.01	0.08	0.25	0.50	1.00	2.00	4.00	6.00	8.00	10.00

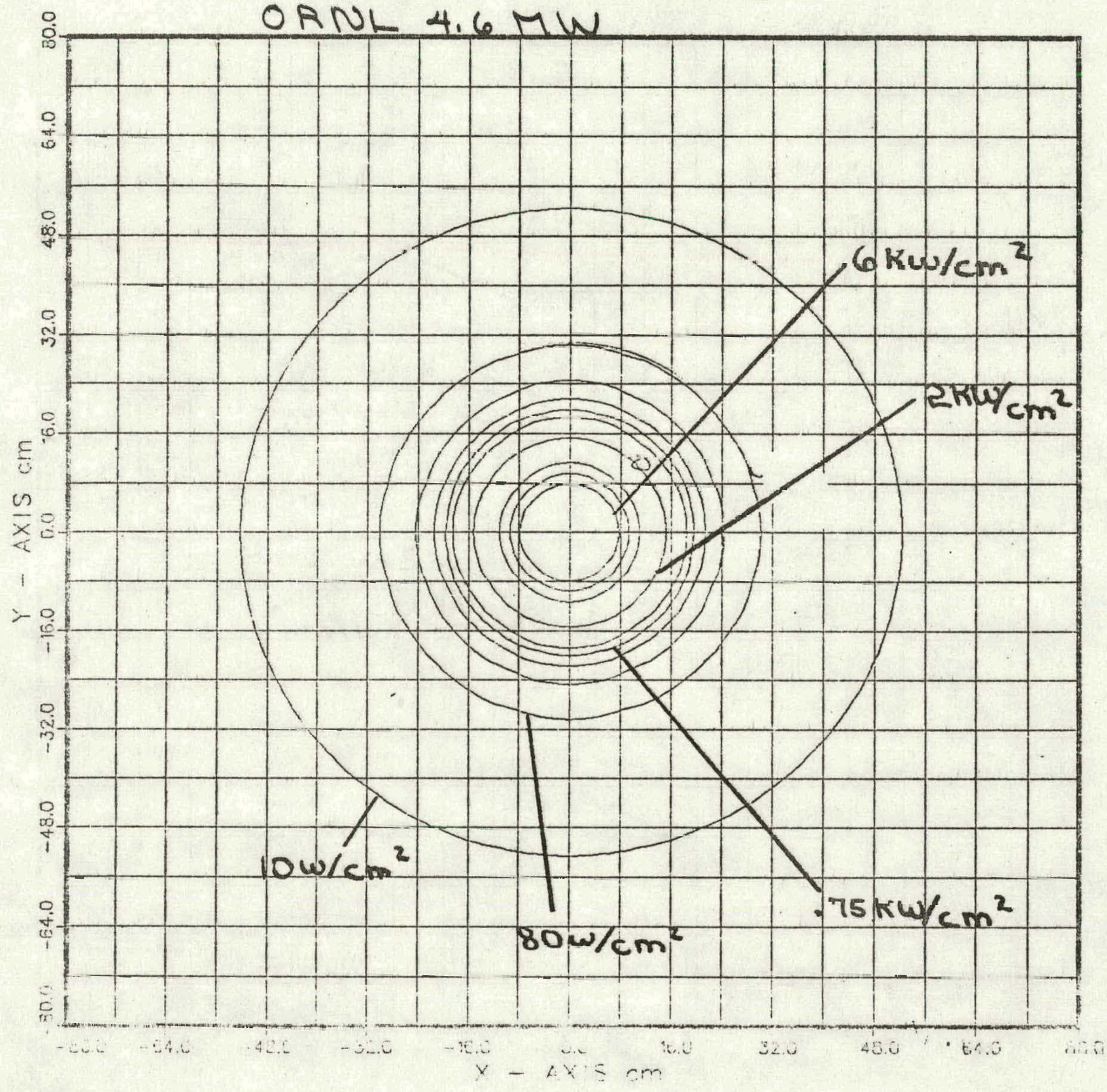
LOCATION 11.5 M FROM SOURCE

POWER DENSITY CONTOUR PLOT

ORNL 13*4.3 .6–90 1.6–10 FX–FY–1300.

SOURCE POWER–kW 4140.00 460.00

ORNL 4.6 MW



BEAMLINE NO. 1	PLOT ORIGIN	0.	0.	1150.
TWO DIM PLOT NO.	PLOT NORMAL	0.000	0.000	1.000
	MAX. POWER DEN.	9.092		

CONTOUR LEVELS	0.01	0.08	0.25	0.50	0.75	1.00	2.00	4.00	5.00	6.00

LOCATION 11.5 m FROM SOURCE.

APPENDIX K

APPENDIX K

OAK RIDGE NATIONAL LABORATORIES RESPONSE

OAK RIDGE NATIONAL LABORATORY
OPERATED BY
UNION CARBIDE CORPORATION
NUCLEAR DIVISION

POST OFFICE BOX Y
OAK RIDGE, TENNESSEE 37830

September 17, 1981

P. Y. Hsu
TPE-II Program Manager
Fusion Technology Program
EG&G
P.O. Box 1625
Idaho Falls, Idaho 83415

Survey of Test Facilities Suitable for Thermal-Hydraulic and/or Thermomechanical Testing of Fusion Blanket/Shield Components

Dear Dr. Hsu:

This letter is in response to your request that we participate in the subject survey (letter Sept. 11, 1981; P. Y. Hsu to O. B. Morgan, Jr.).

ORNL has had many years of experience related to fusion power development and has strongly supported the establishment of a first wall, blanket, and shield verification testing program. In this regard, we responded to the original DOE/OFE request for information concerning the establishment of such a program, we participated in the March 11-13, 1980 workshop at ANL, and submitted a proposal on Jan. 8, 1981 for TPE-I in response to the ANL request for Expression of Interest. As these response documents contain substantial descriptions of ORNLs facilities and capabilities, they are enclosed along with a response to your Enclosure 2 Questionnaire.

We have long been a leader in fusion energy development as evidenced by our ISX (tokamak) program, EBT program (leading alternative to a tokamak), materials programs, superconducting magnet development, the Large Coil Test Facility, Gyrotron Program, and the Neutral Beam Programs. In addition, we have maintained an active liquid metal development program, and have been continuously active in helium-cooled high temperature gas-cooled reactor research. We have had the lead role in developing and supplying neutral beam injectors for the national fusion energy programs. This should be of interest because neutral beam facilities appear to provide an ideal environment and can serve as a heat source for first wall testing.

P. Y. Hsu
Page 2
September 17, 1981

We invite you to visit ORNL and see our facilities firsthand. If you desire such a visit or if there are any questions about this response or need for additional information, please call me at FTS 624-0558.

Yours truly,

T. S. Kress

T. S. Kress
Engineering Technology
Division

TSK:kfr

Enclosure

cc(w/o enclosure): L. A. Berry
H. H. Haselton
T. J. Huxford
R. E. MacPherson
O. B. Morgan, Jr.
J. B. Roberto
M. W. Rosenthal
J. Sheffield
H. E. Trammell
D. B. Trauger
C. D. West

QUESTIONNAIRE FOR TPE-II FACILITY SURVEY

1. What facilities do you have for conducting steady or cyclic thermo-mechanical heating experiments?

   a. What is the heat source?

   b. What is the test environment (vacuum, inert gas, etc.)?

   c. What size and configuration of test pieces will they accommodate?

   d. What materials can be accommodated in the test space?

   e. What range of temperatures and/or power can be achieved?

   f. What are the rise and decay times of temperature/power in these facilities?

   g. What capability exists for active cooling of a test piece in these facilities?

   h. How are these facilities instrumented?

   i. What is the availability and approximate operating cost of these facilities?

   j. What supporting facilities are available, e.g., machine shops, analytical laboratories, etc.?

   k. What are the numbers and qualifications of personnel available to support tests in these facilities?

   l. What other aspects of these facilities would be pertinent to tests of this type?

Response to Enclosure 2 (Questionnaire for TPE-II Facility Survey)

1. Facilities

- The Core Flow Test Loop (CFTL) - a large sophisticated helium circulating facility with associated 4 MW power supplies and PDP-11 data acquisition system.
- A large ultra-high G.E. vacuum chamber
- Miscellaneous water circulating equipment (pumps, valves, piping, etc)
- Neutral Beam Injection Facilities
- The ISX Test Facilities

a. - Neutral Beam Injectors

- Small Plasma Torches
- Direct Electrical
- Radiant heat sources can be obtained from space industries

b. - Vacuum

c. - Size and configuration of test pieces depends on facility. The vacuum chamber is 1.2 M diam. by 3.35 M high and could probably accommodate a test specimen of the order of 5000 cm^2

d. - Any appropriate first-wall materials (e.g. graphite tile, stainless steel, aluminum, etc.)

e. - Unknown

f. - Unknown

g. - Helium and water

h. - Instrumentation will have to be supplied for test pieces. Facilities have miscellaneous instruments for flow, temperature, and pressure measurement.

i. - CFTL, vacuum chamber, water equipment, neutral beam facilities and other ISX facilities have partial availability. On definition of program, facility availability and cost would be determined.

j. - ORNL is a large interdisciplinary laboratory with extensive support facilities (machine shops, chemistry, physics, and materials laboratories, and large computer facilities).

k. - The facilities are controlled by ORNLs Fusion Energy Division and Engineering Technology Division. Consequently, a large cadré of highly qualified and experienced personnel are available from which to choose support personnel for first wall testing.

FIRST WALL DEVELOPMENT

T. S. KRESS

PREPARED FOR FUSION PROGRAM PLANNING MEETING

OAK RIDGE

FEBRUARY 9, 1981

[illegible] OF THE [illegible]

[illegible] THE BASIS [illegible] MECHANICAL [illegible] AND MECHANICAL STRA[illegible] FOR [illegible] DESIGN

[illegible]

[illegible]

[illegible] FRICTION [illegible]

A LITTLE BACKGROUND ON THE FIRST-WALL PROGRAM ELEMENT:

- <u>PRESENTLY ONLY A PROPOSAL</u>

- STARTED WITH LETTER FROM COFFMAN ANNOUNCING DOE/OFE'S INTEREST IN STARTING A FIRST-WALL PROGRAM AND REQUESTING INFORMATION ON CAPABILITIES

- ORNL TEAM SUBMITTED A PROPOSAL

- ANL SELECTED AS LEAD LAB WITH GUIDANCE THAT THEY MUST INVOLVE OTHER LABS, UNIVERSITIES AND INDUSTRY

  ANL SET UP A WORKSHOP TO HELP DEFINE THE PROGRAM WHICH WAS DIVIDED INTO FIVE SEPARATE PROGRAM ELEMENTS (ORNL PARTICIPATED)

- AS A RESULT OF WORKSHOP, ANL ISSUED REQUEST FOR "EXPRESSIONS-OF-INTEREST" FOR SUB-CONTRACTING THREE OF THE FIVE PROGRAM ELEMENTS. WE ISSUED ABOVE PROPOSAL FOR ONE OF THE PROGRAM ELEMENTS - "FIRST-WALL THERMAL HYDRAULIC/THERMOMECHANIC TESTING"

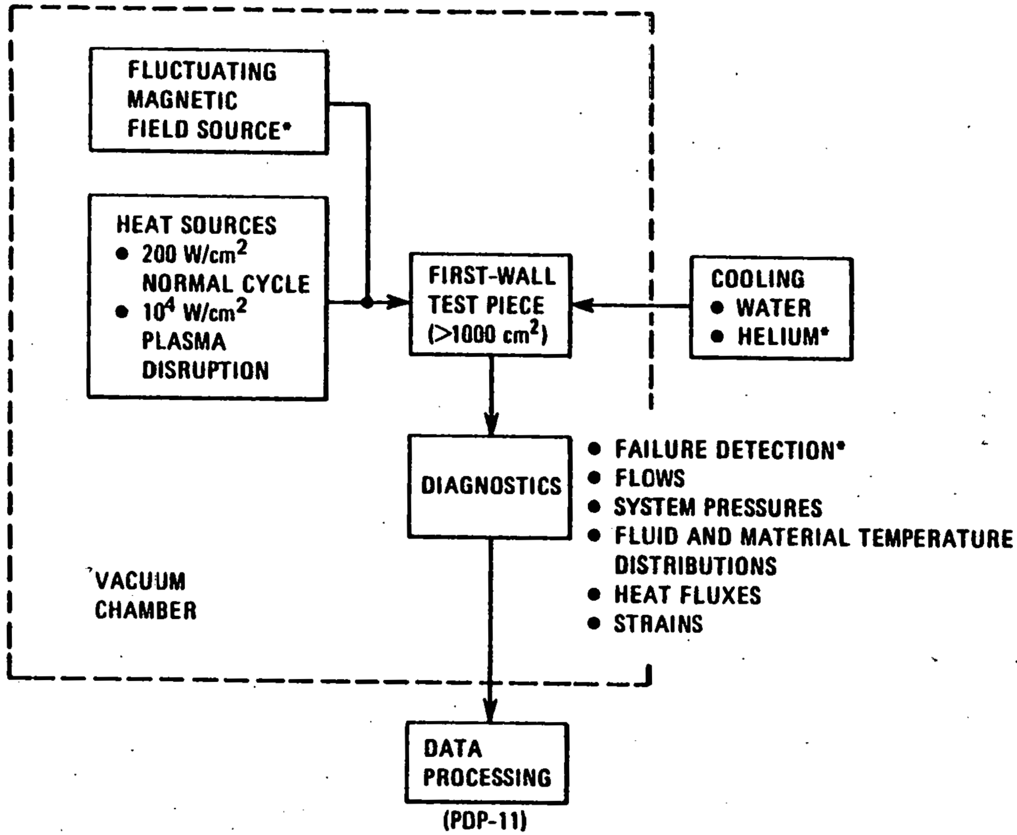
THE OBJECTIVES OF THE FIRST-WALL PROGRAM ELEMENT ARE:

- TO PROVIDE THE BASIC HEAT TRANSFER, FLUID MECHANICS, AND MECHANICAL STRAIN PERFORMANCE DATA FOR FIRST WALL DESIGN
  - ASSYMETRIC SURFACE AND VOLUME HEATING
  - CLYCIC LOADING
  - WATER COOLED TUBULAR DESIGNS
  - LIMITERS
  - COOLED AND UNCOOLED ARMOR
  - HELIUM COOLED MODULAR DESIGNS
  - PLASMA DISRUPTIONS

- TO PROVIDE OPTIMIZATION AND QUALIFICATION TESTING FOR DESIGN OPTIONS
  - FAILURE MODES AND FREQUENCIES
  - LIFETIME PROJECTIONS
  - COMPUTER MODEL DEVELOPMENT AND VALIDATION

SIMPLIFIED SCHEMATIC DIAGRAM OF PROPOSED FIRST-WALL THERMAL HYDRAULIC/THERMOMECHANIC TEST FACILITY

ORNL-DWG 80-3690 FED



*POSSIBLE PHASE-II COMPONENTS

Figure A-1

DRAFT

Contract No. W-7405-eng-26

FIRST WALL, BLANKET, SHIELD (FWBS) TEST PROGRAM

July 1979

OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37830
operated by
UNION CARBIDE CORPORATION
for the
DEPARTMENT OF ENERGY

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1. INTRODUCTION

Oak Ridge National Laboratory (ORNL) is in total support of a development program that would provide experimental testing capability for first wall, blanket, and shield (FWBS) components. We perceive that such a testing program will be a crucial element of the national program as fusion moves into a more reactor-like phase.

The need for such a program arises because of the unique combination of thermal, structural, neutronic, magnetic, and materials problems that are to be encountered in reactor-like FWBS systems and because of the fact that there is only limited technological experience that is directly applicable.

In this document we propose an aggressive program that would provide direct support to the FWBS needs of devices such as the Engineering Test Facility (ETF), the Experimental Power Reactor (EPR), and the Commercial Demonstration Reactor (DEMO) as they become part of the national program.

In such a program, there appears to be a considerable amount of valuable testing that can be done using existing or easily developed facilities. Consequently, we recommend initiation of this program with <u>separate effects experiments</u> that are relevant to a variety of the designs described in the paper studies. The advantages of such an effort are as follows.

1. Testing could begin in FY 1980.
2. Utilization of existing facilities would have the least impact on OFE's budget.

3. Experience necessary to define a dedicated FWBS test facility would be obtained.

4. Crucial design data would be available for ETF FWBS preconceptual design efforts.

To confidently provide FWBS hardware for ETF, EPR, and DEMO, a dedicated test facility will have to be constructed to develop and qualify FWBS components and complete assemblies. The inherent complexity of FWBS systems coupled with the difficulty of doing routine maintenance on such systems will require that they be very reliable. The only way such reliability can be achieved is by incorporating testing that simulates the reactor environment as faithfully as possible. A dedicated FWBS facility as described in this document appears capable of meeting these needs.

We believe the most effective FWBS verification testing program will result if ORNL is given the lead role in conducting and directing the program. The following considerations, included in the text of the proposal, support this conclusion.

1. As a result of ORNL's participation in a variety of programs, significant equipment and capability that are directly applicable already exist.

2. The ongoing strong participation of ORNL in the national fusion energy program provides the opportunity for close coordination, particularly with ETF, and responsiveness to DOE needs.

3. ORNL management has consistently demonstrated unqualified support of the fusion energy program.

4. ORNL has already demonstrated the ability to incorporate industry and outside contract participation.

2. FWBS TEST PROGRAM GOALS AND OBJECTIVES

The major goals and objectives of the proposed FWBS Test Program are as follows:

1. To provide direct support to ETF or similar programs by establishing an Engineering Design Data Base for FWBS systems;
2. To provide a dedicated testing facility in order to qualify FWBS systems prior to their installation at the ETF site;
3. To provide relevant FWBS experiments during ETF operation so that subsequent EPR development needs can be identified and accommodated; and
4. To provide, in addition to direct testing and development support, recommendations to OFE for the development needs of parallel programs necessary to implement EPR and DEMO FWBS systems.

3. RECOMMENDED FWBS VERIFICATION TESTING PROGRAM

3.1 GENERAL DESCRIPTION OF PROPOSED TEST PROGRAM THROUGH ETF

In initiating the test program, the various paper studies would establish the basis for conducting experiments. In this context, the initiation of this test program must be preceded by a technical audit of any design or design feature that is allegedly ready for development testing. The major technical uncertainties should be identified in such an audit. This procedure will determine whether there is a need for more analysis and design and define the nature of specific experiments.

In a broader sense, this procedure should be adopted throughout all phases of this test program. The major point is that a development testing program requires a continuing and coordinated effort involving analysis, design, and experiment.

The discussion that follows describes the test program in terms of separate elements that refer to the specific scope of the testing to be performed. The chronology of actually implementing the test program would probably involve considerable overlap between these elements.

Element I — Generic Research and Development (R&D): Separate effects experiments, to be conducted in a variety of existing (or easily developed) facilities will address the basic issues concerning whether the design approaches used in the paper studies are verifiable within available technology.

Element II — Prototype Component Testing: When the generic R&D testing has established the validity of specific concepts, the program would justify the fabrication of full- or subscale FWBS components to undergo simulated reactor testing. Such testing would involve performance and endurance evaluation in a versatile, dedicated facility capable of simulating the combined effects of the reactor environment.

Element III — Integral Assembly Proof Testing: The scope of this testing effort will be to verify the performance and reliability of the integral blanket, shield, and first wall assembly, including all the necessary piping and support structure. Ensuring that FWBS systems are qualified, insomuch as is possible, before delivery to the ETF site will be the intent of this testing, to be performed at the dedicated site developed for Element II.

3.2 ELEMENT I — GENERIC R&D

3.2.1 First Wall Generic Testing

In the absence of divertors, it is estimated that the surface heating rates on reactor-like first walls is in the range of 50–100 W/cm^2. In addition, intermittent surface heating rates as high as 2.0×10^4 W/cm^2 for roughly 20 msec or so could occur as a result of a major plasma disruption.

A review of the paper studies indicates various distinct design approaches for accommodating this first wall heat flux. Arrays of small diameter tubes, annular passageways interspaced between pressure vessels, and large irregular flowing liquid metal coolant passages predominate current conceptual design approaches.

The following important facts characterize the first wall heat flux problem and the designs that have been suggested to accommodate its effects:

1. The number of cycles for reactor-like device is 10^5 to 10^6. It is generally regarded that the first wall must be cooled with flowing fluid, must be leak-tight to vacuum standards, and must be metallic. A search for an application in which all these conditions are met appears to reveal that there is no precedent.
2. Analyses of proposed first walls have been, in general, approximate and cursory, and it can be expected that the details of thermally induced stresses will be worse than shown by analyses.

In consideration of the above factors, it is suggested that the most urgent testing requirement insofar as first walls are concerned is to establish experimentally that this difficult area is not beyond the limits of the proposed designs or existing technology.

The scope of the testing at the generic level would identify the first wall geometry and heat flux as the primary experimental variables. Other experimental features, such as test temperature, materials, test atmosphere, coolant purity levels, etc., would be selected for convenience.

In order to accomplish this first wall generic testing, the use of existing coolant loop facilities with appropriate modifications appears the most practical option. These modifications would provide a heat flux capable of simulating the correct magnitude, duration, and number of cycles.

A relatively simple method of simulating the heat flux during normal plasma operation would be to use arrays of tungsten filament quartz lamps with planar reflectors. These lamps have been available commercially with flux capabilities up to 215 W/cm^2. The plasma disruption heat flux simulation could be provided by intense heating devices such as a plasma or Marshal gun or a neutral beam injector that could provide not only the right magnitude but also the very short duration pulses that are felt during a major plasma disruption.

It is expected that this testing will provide design data helpful in the optimization of first wall performance through structural and thermal hydraulic innovations. In addition to the value to blanket first walls, such testing would be of considerable value to other ongoing programs. For example, some of the severe problems facing the development of limiters, divertor particle collectors, and ancillary hardware typical of long pulse neutral beam injectors could be approached through the use of the test data and apparatus.

3.2.2 Blanket and Shield Heat Sink Generic Testing

Paper studies related to these components have suggested various schemes to remove the heat internally deposited in the moderator and shielding materials. The possible effects of calculational errors associated with these paper study designs are, in concept, not as serious as those associated with the first wall designs. This situation arises primarily because of the relatively low power densities in the blanket and shield components. In addition, there appears to be ample design capability for providing more than adequate surface cooling areas to the heated volumes of the various blanket and shield concepts.

There is, however, some concern for specific types of designs. In particular, some of the solid breeder designs where the coolant is passed over moderator beds constructed of blocks, pebbles, or and powders need generic testing. In these cases, there is the potential for ~~very~~ deleterious stagnation regions occurring in the coolant flow path. Because of this, tests should be provided either to verify or develop ways of preventing such phenomena. This type of testing could probably be accomplished by bench-type fluid mechanics tests and would not require simulation of internal heating of the moderator beds; i.e., a detailed description of the flow path would adequately verify the heat transfer mechanisms.

3.2.3 Generic MHD-Related Testing

In considering blankets where liquid lithium is circulated as the coolant as well as blankets where the breeding material is liquid lithium not circulated for cooling, MHD aspects must be considered. Within the past seven or eight years, nearly all of the advances made in understanding these interactions for fusion-related goals have been funded on a relatively small scale. If this material is to be seriously considered for FWBS use, the level of understanding of these phenomena needs to be improved. The following areas need improvement in levels of understanding that can be obtained by generic testing:

1. Circulation induced by thermoelectric effects that arise because of the dissimilar nature of lithium and containment materials;

2. Improvement in the state of knowledge of flow resulting when lithium is circulated in small tubes that are themselves immersed in lithium;
3. Extent and consequences of stagnant regions as they affect coolant flow and tritium removal; and
4. Effects of changing poloidal field on fluid motion and stresses.

To provide adequate experimental capability to match FWBS conditions requires: (1) large test volumes, (2) high magnetic fields, and (3) capability for flowing large volumes of liquid lithium. The first two items are already well satisfied at ORNL in the form of the Magnetic Isotope Separation Facilities (MISF) in which a volume 1.2 m diam by 0.3 m thick can presently provide 0.65 T on a steady-state basis (see photographs of facility in Appendix). If the lithium flow capability is added to this facility, the important dimensionless parameters can be increased by about one or two orders of magnitude over what has previously been available with smaller magnets and with mercury as the operating fluid.

To make this picture more attractive, implementation of this capability would, it is anticipated, provide a key factor for the expected divertor technology program at ORNL. The need here is to deal with lithium flow as a divertor ion-and-energy collection medium.

3.2.4 Materials Compatibilities

From the paper studies, likely breeder/structure/coolant combinations have been identified. The compatibility of these materials with each other within the typical reactor environment can be established in standard metallurgical testing systems that would be designed to:

1. Determine temperature and impurity limits for lithium compatibility with austenitic stainless steels, selected higher nickel alloys, and selected vanadium and titanium alloys;
2. Determine chemical compatibility of selected solid breeding materials with "commercial" helium;
3. Determine compatibility limitations of selected structural materials with commercial helium;
4. Determine compatibility of graphite cladding materials.

The issue of activation specie transport involves species such as ^{60}Co generated in the first wall region that would be transferred to the blanket moderator by liquid/metal corrosion. Such products would then be transported to components outside the reactor such as the primary heat exchangers or the tritium extraction equipment. These phenomena can be appropriately studied in in-pile facilities such as exist in the Oak Ridge Research Reactor (ORR), where there are access holes approximately 0.5 x 0.6 m to the reactor face that can accommodate a variety of in-pile experiments. In addition, stable isotopes can be used to make such studies in the materials compatibility tests.

3.3 ELEMENT II — SUBSYSTEM COMPONENT TESTING

3.3.1 A Dedicated FWBS Test Installation

The intent of this testing will be to verify the performance and reliability of FWBS components at the module or subassembly level.

During such testing, it will be of crucial importance to simultaneously expose representative full- or subscale specimens to the combined effects of the reactor environment. In order to accomplish this, the test environment would be required to match the magnetic, vacuum, temperature, etc., conditions of the designs being tested as closely as possible.

Experiments at this level appear to require a special, dedicated thermal/hydraulic/mechanical test installation. Such an installation would have to be versatile enough to accommodate different design concepts. In the discussion that follows, a description of the testing and facility requirements will be given.

3.3.2 Thermal/Hydraulic Considerations during Subassembly Qualification Testing

Thermal/hydraulic simulations will include surface heating (normal and disrupted) and internal heating. The testing would be performed using the actual coolant circulating at a rate and pressure necessary to provide the temperatures, temperature gradients, and pressure-induced stress fields as designed. The testing would probably be done in a vacuum environment to simulate material~~s~~ fatigue response and to allow identification of failures characterized by small vacuum leaks in the coolant circuits. The facility will be required to provide this simulated environment for a duration typical of the 10^5–10^6 reactor cycles. Various coolant loops to provide gases, molten salt, liquid metals, and pressurized water will have to be provided that are capable of absorbing several megawatts.

3.3.3 Materials Considerations during Subassembly Qualification Testing

During this testing phase, materials considerations will require considerable attention if valid verification of components is to be expected. In particular, the test specimen structure must be made of candidate alloys. The specifications, such as degree of cold work, surface finish, welding practice, and the like, must realistically match the intended application. Likewise, the materials environment involving coolant and moderator impurity levels and operating temperatures must also match the designs being tested.

Although it might be desirable to perform this testing on highly irradiated test specimens, the cost and uncertainties of such tests discourage their use. An alternative approach would be to modify the composition or heat treatment of blanket modules to simulate the radiation-induced loss of ductility. These modifications would be relatively minor and would be done in such a way that the failure mode would be that of the irradiated material. The principal advantage to this approach is that the specimens can be tested without expensive shielding, remote handling, and the like. Furthermore, the test volume required to irradiate typical FWBS components is such that the observed neutron fluence could not be achieved in a reasonable time.

3.4 ELEMENT III — INTEGRAL ASSEMBLY TESTING

Even after a successful component development program, the inherent complexity of FWBS systems will necessitate the qualifications of complete FWBS assemblies. This effort will require testing apparatus in which these representative assemblies will undergo both development and eventual proof testing. Here tests will be performed that are critical to the assembly performance and reliability. Some testing may require providing reactor-like thermal and magnetic environments for the assembly. Some typical testing would be:

1. Verification or development of coolant manifolding that will ensure proper flow distribution to the various FWBS components;
2. Verification or development of schemes to provide adequate thermal isolation between entrance and exit coolant streams without deleterious structural effects;
3. Verification or development of startup schemes for liquid metal and/or molten salt coolants and/or moderators;
4. Verification or development of leak detection methods for all coolant connections or passages exposed to critical vacuum; and
5. Verification that design of FWBS support structure is adequate to sustain dead weight and magnetic and thermal loading.

In a parallel effort, the remote maintenance development (welding, cutting, handling) that will be necessary for a full-scale machine using a particular blanket concept should be under way. Full-scale modeling activities will determine the final practicality of a design.

3.5 RELATIONSHIP TO EPR AND DEMO

The specific goals of the test facility for an EPR first wall, blanket, and shield depend so heavily on the outcome of ETF testing that only general comments can be made at this time.

Certainly from ETF operation, basic source data will be obtained that will describe the actual dynamics to be anticipated in power reactor operations. In addition, operational experience will be obtained with which to weigh the relative merits of the various test modules introduced into ETF. From this data, the FWBS test facility would be updated, as necessary, to do qualification and development testing for the EPR FWBS system. In addition, the test facility could be modified to incorporate the balance of plant ancillary equipment necessary for an EPR. In general, the test program for an EPR would probably proceed in a way similar to that of ETF with the inclusion of specific development needs of coordinating the FWBS system with steam generators, turbines, maintenance procedures, tritium recovery and processing systems, etc. The goals of the test facility would also involve the development of systems that are optimized to reflect the needs of commercial power generation.

4. ORNL CAPABILITIES AND FACILITIES

4.1 FWBS TEST PROGRAM SUPPORT

For thirty years, ORNL has been engaged in activities that have led to the development of expertise pertinent to fusion reactors. The activities of the Fusion Energy Division are well known, but other capabilities and available facilities may be less apparent. ORNL has had an active liquid metals development program during this entire period, supporting initially the Aircraft Nuclear Propulsion Program, then the Space Program, and now the Liquid Metal Fast Breeder Reactor. This has led to a wealth of experience in liquid metal systems design and operation, handling and safety, materials corrosion characteristics, instrumentation, components, and special requirements. Several major liquid metal facilities are currently in operation, and a sodium thermal shock facility is in standby. We have been continuously active in support of the helium-cooled High Temperature Gas-Cooled Reactor and are currently supporting the Gas-Cooled Fast Breeder Reactor by building a major helium-cooled test facility. Thus, there is a familiarity with helium technology paralleling that in liquid metals.

Two research reactors, the High Flux Isotope Reactor and the Oak Ridge Research Reactor are available for simulated irradiation tests of pertinent materials. Several large buildings (typically 90 x 90 m, three stories) occupied by ORNL in the Y-12 Plant are in use as office and test space. These buildings have crane-equipped high bay areas (typically 30 x 90 m, 15-m ceiling), appropriate utilities, and extensive additional space that is currently available for test facility installation.

In addition to the above, those facilities and equipment at ORNL that appear to be directly applicable to the verification testing program include the following.

1. The Magnetic Isotope Separations Facility: There exist some presently operable unique facilities built during World War II as electromagnetic isotope separators. These units contain large coils capable of producing a magnetic field of 0.65 T uniformly across a 0.4 m gap in a normally evacuated test space of dimensions approximately 1.2 m diam by 0.3 m thick. Such a facility could serve, initially, to investigate the MHD effects and magnitudes of cyclic magnetic loadings on particular module geometries. It would be an ideal base on which to develop the dedicated FWBS test installation by adding the appropriate heat sources and lithium/helium/water systems.

2. The Large Coil Program Facility (LCPF): The ORNL magnet lab will have a toroidal volume capable of about 4 T on axis. This facility is expected to be used through 1983 to complete the testing of the large coils. The LCPF power supplies, cryogenics, etc., as well as the chamber itself, may be used after that date to simulate the magnetic effects of a major plasma disruption.

3. Neutral Beam Injectors: ORNL has had a lead role in developing and supplying neutral beam injectors for the national fusion energy program; consequently, the capability and experience, as well as existing devices, are already on hand. In addition to being a possible heat source to simulate a plasma disruption, a neutral beam

injector may be a convenient source of hot particles that can be used in experiments to establish the feasibility of proposed divertor particle- and energy-collecting schemes. They can also be used to investigate material sputtering due to hot particle impingement on the first wall.

4. Materials Testing Loops: As a result of ongoing and past programs at ORNL, there are a variety of sodium, NAK, helium, and molten salt test loops and equipment in different stages of development. These should be investigated to determine the most appropriate for reactivation and use as a lithium/helium/structure materials compatibility test system and for use as heat transfer tests.

In addition to the availability of many items of specialized equipment, ORNL has a unique interdisciplinary capability in the combination of a strong fusion energy activity, a sophisticated materials and fabrication development capability, and an experienced systems- and component-testing group backed by a flexible engineering staff. ORNL also has ready access to a professional staff possessing the following applicable experience and capabilities:

1. Design, construction, and operation of complex test facilities;
2. Liquid-metal-testing and handling technology;
3. Strengths in experimental engineering, instrumentation and controls, heat transfer and fluid mechanics, and solid mechanics (fracture, fatigue, creep, crack growth, stress, strain, and familiarity with finite element techniques);
4. In-pile irradiation testing;

5. Ability and techniques to simulate irradiation damage;

6. Remote welding and cutting experience and apparatus.

4.2 JUSTIFICATION FOR SELECTING OAK RIDGE AS THE SITE FOR THE VERIFICATION TESTING PROGRAM

Although portions of the generic R&D activities could be placed at a variety of sites, there are obvious advantages and powerful incentives for the overall verification testing program direction and management to reside at an interdisciplinary national laboratory. ORNL can provide:

1. Existing capability and experience with
   - liquid metal technology,
   - helium technology,
   - fusion technology,
   - MHD,
   - structural mechanics,
   - in-pile irradiation;
2. Availability of directly applicable facilities or apparatus:
   - liquid metal and helium test facilities,
   - very high heat flux sources (neutral beam injectors, plasma guns, quartz lamps),
   - magnetic-field-generating equipment,
   - irradiation damage simulation techniques,
   - artificial crack-initiating techniques,
   - in-place computer codes for heat transfer, fluid mechanics, and finite element structural mechanics,
   - instrumentation and controls,
   - data acquisition systems;

3. Access to a multidisciplinary professional staff and laboratories;
4. Potential for close coordination with ETF and other tokamak experiments;
5. Adaptability and responsiveness to DOE needs;
6. Availability of support facilities (shops, laboratories, assembly areas, I&C, computers, chemical and metallurgical laboratories, etc.);
7. Dedicated, informed, and committed program managers.

5. PROGRAM EXECUTION

5.1 INTRODUCTION

We believe the program as recommended in this document is responsive to the needs for verification testing of FWBS design concepts. The scope includes assessment of proposed design concepts, support and generic testing to assess the feasibility and to optimize the design concepts, and the development and installation of a dedicated FWBS test facility to provide overall integral verification to FWBS modules of particular designs before they are committed for further testing in an ETF. The generic program as outlined makes maximum use of existing capabilities to provide fundamental data on:

1. First wall and blanket heat transfer and fluid flow
2. MHD effects due to interactions with transient magnetic fields and
3. First wall, blanket and shield materials compatibility.

In addition, the dedicated test facility will provide endurance, performance, and maintainability verification for specific concepts under a simulated reactor environment that includes:

1. Vacuum on the "plasma" side;
2. Simulated plasma heat sources;
3. Appropriate transient and steady magnetic fields; and
4. Appropriate first wall, blanket, and shield materials including simulated irradiation damage for the first wall.

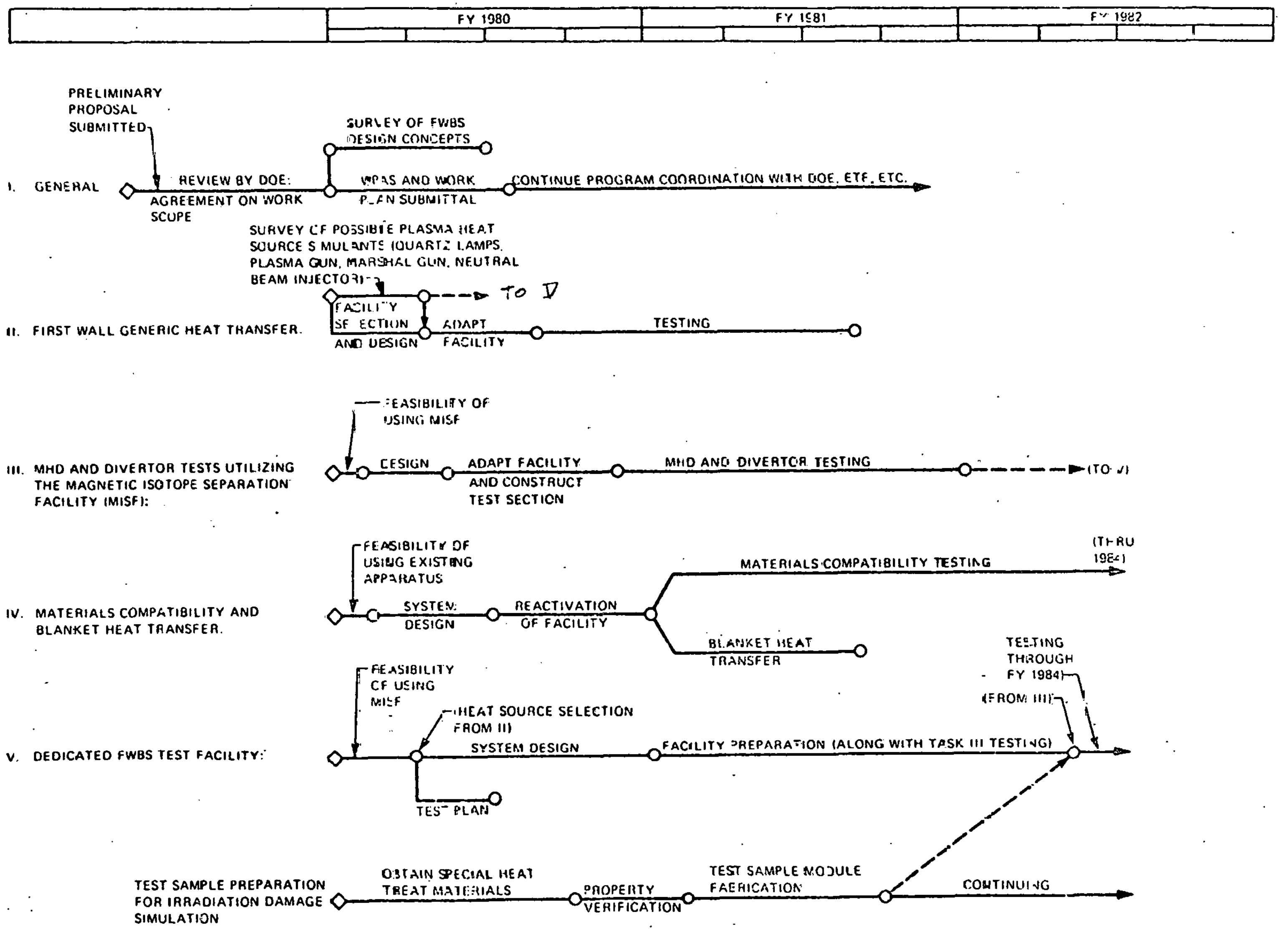
We propose to initiate this program by appointing an experienced program manager from the Engineering Technology Division. A team would be assembled along with a committee to review and perform technical audits of FWBS design paper studies. The objectives of the audits would be to evaluate the overall applicability of the designs, establish specific technical uncertainties and risks, and recommend testing to quantify the uncertainties and reduce the risks. Partially based on the audits, a complete program plan and work breakdown structure (WBS) would be developed. Existing facilities and equipment would be surveyed to establish their appropriateness and potential for use in the testing program. These would primarily consist of: The Magnetic Isotope Separations Facility; The Large Coil Program Facility; various heat sources (plasma gun, Marshal gun, quartz lamps, neutral beam injectors); and existing materials and heat transfer test loops.

Such a program can be started in FY 1980 if agreement is reached and the program supported by DOE/OFE. A tentative schedule based on that assumption is provided in Section 5.3, below.

5.2 LEVEL OF EFFORT

At the writing of this document, we have had insufficient time to prepare realistic and reliable estimates of the yearly costs and manpower requirements for the proposed verification testing program. We will be prepared to discuss these in later correspondences or meetings.

5.3 FWBS VERIFICATION TESTING PRELIMINARY ACTIVITY NETWORK AND SCHEDULE



ORNL Proposal for TPE-I of the First Wall/
Blanket/Shield Engineering Test Program

T. S. Kress
T. J. Huxford
R. E. MacPherson

January 8, 1981

ORNL Proposal for TPE-I of the First Wall/
Blanket/Shield Engineering Test Program

Introduction

We view the establishment of a strong first-wall testing and qualification program to be of critical importance and of high priority at this stage of the national fusion energy effort. Consequently, we eagerly responded to the original DOE/OFE Request for Information (letter: F. E. Coffman, June 5, 1979), participated in the subsequent workshop (Washington, March 11-13, 1980), and are now prepared to commit significant resources and talents to the program.

ORNL believes it can make major contributions to a first-wall engineering testing program, especially in the area defined as Test Program Element-I (TPE-I). This proposal, then, is our response to the November 24, 1980 Request for Expression of Interest (EOI). Our response, that follows, is structured by lettered headings that correspond respectively to the list of items requested in the EOI letter with supplementary information as appendices.

A. Approach

In planning for a test program that will provide a first-wall thermal hydraulic/thermomechanic engineering data base, it must be recognized that the capability is required for testing with both water and helium as possible primary coolants. A water-loop facility can usually be assembled relatively inexpensively. However, a significant high-pressure helium circulating facility (with an associated data acquisition system) can prove to be costly and time-consuming to design, construct, and assemble. Consequently, it would be desirable to start from the base of an existing helium facility that has the desired capabilities.

Although much applicable thermal hydraulic data can be developed in an open environment, it is also preferable (ultimately essential) that such testing be conducted in an appropriate vacuum.

Consequently, with the above considerations in mind, an approach for TPE-I is proposed that would make optimum use of existing facilities and equipment, provide early meaningful results in support of the Fusion Engineering Design Center (FEDC; formerly ETF), and provide the flexibility for easy expansion into a broader-based first-wall thermal hydraulic/thermomechanic testing facility. The essential elements of this approach at ORNL are outlined as follows:

- Re-activation of an existing large vacuum chamber now in standby status (see Section H and Appendix B), making any modifications needed to conform to the FW/B/S TPE-I requirements;

- Utilization of existing water-loop equipment (piping, pumps, heat exchangers, valves, flowmeters, etc.) to assemble a system with the appropriate water cooling capabilities;

- Co-location of the above vacuum chamber and water loop with an extensive helium circulating facility now nearing completion (see Section H and Appendix C). This would allow utilization of the helium facilities PPD-11 based data acquisition system and provide easy capability and accessibility for helium cooling testing;

- Establishment, as a supporting participant, particular aero-space industry with equipment and experience in constructing graphite or quartz filament radiant heaters to provide the required radiant heat sources;

- Investigation of the use of laser beams, arc-plasma guns, and neutral-beam injectors as alternative heat sources to provide the capability of assessing plasma disruption effects.

With this approach, a flexible test facility will be established at minimum cost that has the capability of conducting tests in a vacuum environment with either water or helium cooling. The availability of the sophisticated high speed data acquisition system ensures the ability to take extensive data even with rapid transients that might result from plasma disruptions or from loss-of-cooling accident conditions. Initial tests will use water as the coolant and will focus on normal cycle thermal effects in support of the FEDC. Key first-wall design data (temperature distributions, heat transfer coefficients, flow distributions, strain levels, etc.) will be established for different wall geometries, cooling conditions, and structural restraints. The system would also be capable of assessing plasma disruption effects and off-normal conditions related to reduced cooling, uneven coolant distributions, and distorted geometries.

The extension into Phase-II could consist of:

- Incorporation of a hot ion source;
- Invoking the helium cooling capability;
- Perhaps providing capability for simulating the magnetic loadings;
- Utilizing the expertise and capability existing in the ORNL Fusion Energy Materials Program to provide test pieces that appropriately simulate the properties at any desired stage of irradiation damage;
- Utilizing ORNL's capability in non-destructive testing to develop the capability of detecting crack initiation and growth and to provide failure detection capability;

Phase-II testing would then include evaluating specific design options in the appropriate environments. These design options would be optimized and qualified for the desired lifetime (>10^5 cycles) with the identification of the potential failure nodes, failure locations, and failure frequencies.

In parallel with Phase-I and Phase-II experimentation, there would be a complementary analytical effort that would evaluate and correlate the experimental results and provide the models and calculational tools required for first-wall design and assessment. These models would consist of a coupling of existing thermal-hydraulic (heat transfer and fluid mechanics) models with structural mechanics models to establish detailed transient temperature distributions and thermal stresses under the cyclic loading conditions. These would provide preliminary design tools needed to optimize and evaluate competing first-wall design concepts. The extension of these to provide the capability of predicting failure locations and frequencies will require the incorporation of new models that would address fatigue crack initiation and growth taking cognizance of the cumulative irradiation damage effects.

Figure A-1 shows a simplified schematic diagram of the proposed first-wall TH/TM system for TPE-I (Phases I and II) and Figure A-2 shows a simplified activity structure for this approach.

B. <u>Incorporation of Fusion Specific Physics, Engineering and Materials Considerations</u>

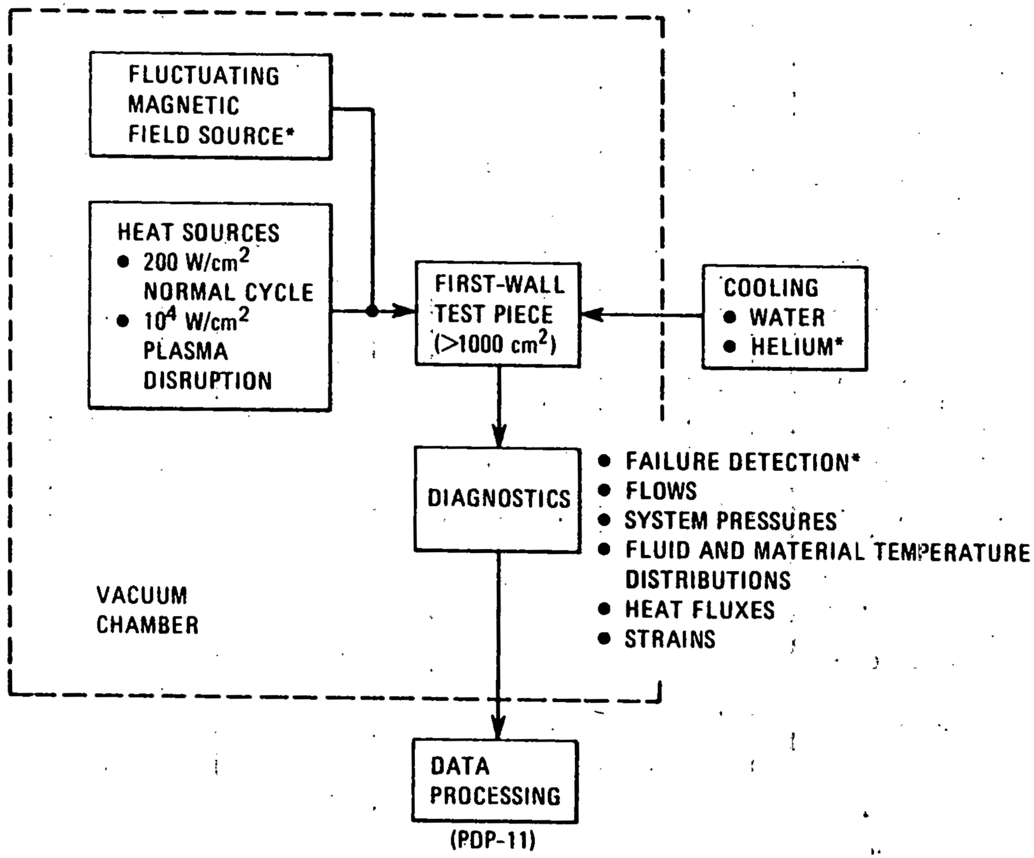
It is clear that the elements that drive the first-wall R&D needs include fusion specific physics (e.g. reactor power cycles, plasma behavior, heat loads, neutron irradiation, plasma disuptions, runaway electrons, etc.), specific engineering designs (e.g. the proposed FEDC water cooled tubular arrangement with its armor plating and limiter and the variety of modular design concepts for prototype reactors), and materials considerations (e.g. plasma/surface interactions, neutron irradiation damage, and constitutive properties related to fatigue resistance, crack initiation and growth). For a relevant first-wall program, it is absolutely essential that these considerations be properly incorporated into the planning and testing and that they be kept current as a continuing part of the program.

ORNL is fortunate to have in-house the expertise base and program structures that will accommodate this need. The required resources and expertise base abound in the various fusion energy activities associated with the ISX programs, the Fusion Engineering Design Center (formerly ETF), the EBT-P Program Office, the neutral beam injector program, plasma/surface interaction studies, fusion materials studies, the Large Coil Test Facility, etc.

The First Wall TPE-I Program would be structured to take maximum advantage of this in-house expertise resource as well as to utilize the

SIMPLIFIED SCHEMATIC DIAGRAM OF PROPOSED FIRST-WALL THERMAL HYDRAULIC/THERMOMECHANIC TEST FACILITY

ORNL-DWG 80-3690 FED



*POSSIBLE PHASE-II COMPONENTS

Figure A-1

ACTIVITY STRUCTURE FOR PROPOSED FIRST-WALL TH/TM
TEST PROGRAM ELEMENT

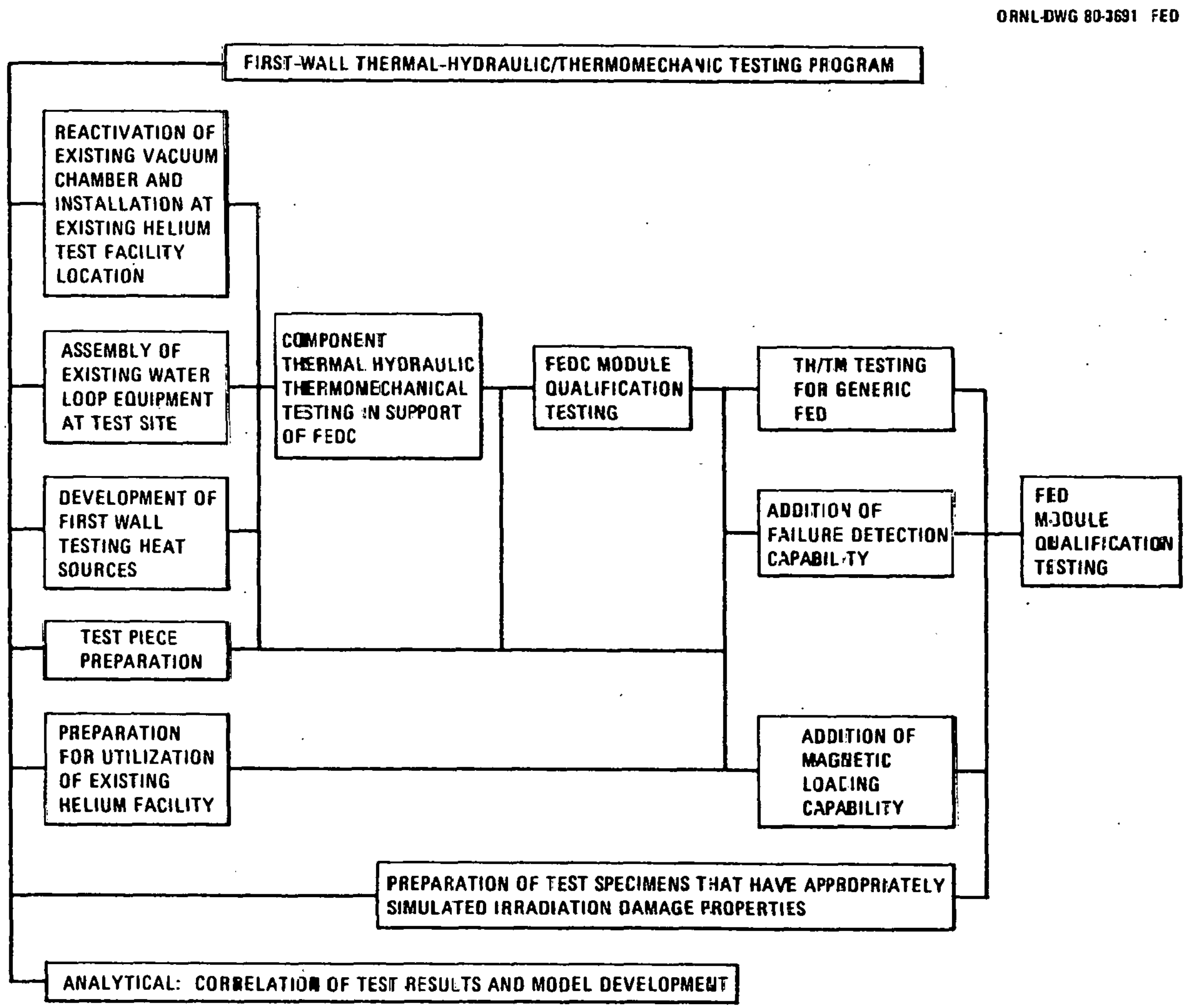


Figure A-2

general interdisciplinary strengths of ORNL. The program would be administered and conducted through ORNL's Engineering Technology Division (ETD) (formerly the Reactor Division) working closely with the Fusion Energy Division. The ETD has had a long history of successful large scale engineering testing associated with fission reactors (sodium cooled breeders, LWRs, molten-salt fueled reactors, gas (helium) cooled reactors, and space power applications involving eutectic NaK and potassium (K)). Therefore, ETD has the expertise, experience, capability, and equipment needed for a superior first-wall program.

To incorporate the required inter-divisional and inter-disciplinary interactions, a matrix program structure is anticipated as illustrated by Figure B-1. As shown in Figure B-1, the proposed First-Wall TPE-I can be divided into various sub-activities that would involve participation and responsibility from various programs and divisional institutions at ORNL (only a few of the identified participants would be 100% on the program).

An internal review committee would be established, made up of carefully selected members (see for example Section F) from the various fusion related activities at ORNL. This committee will provide the desired interactions and continued guidance in areas of plasma physics, plasma/materials interactions, divertor and limiter technology, first wall engineering (FEDC, EBT-P, and general), fusion materials, structural mechanics, and magnetic systems.

The direct line personnel already identified (see Section F) for the TPE-I Program have expertise related to first wall engineering and design, heat transfer, fluid and structural mechanics, and development of complex engineering test facilities. The project manager and other lead participants will maintain active participatory interactions and communication with the general fusion energy programs and in particular, with the activities of the Fusion Engineering Design Center and the EBT-P Program Office - both physically located at ORNL. Strong ties would be maintained with the General Engineering design teams responsible for the FEDC design concepts.

A strong connection would be established to the fusion materials program activities in ORNL's Metals and Ceramics Division to ensure incorporation of relevant materials considerations. It is anticipated that this connection will have major input in establishing test piece construction techniques that will appropriately simulate the irradiation-damage properties of first-walls. The knowledge techniques, and capabilities already exist there for that purpose.

Figure B-1

MATRIX ORGANIZATION INTERACTIONS FOR PROPOSED FIRST-WALL TH/TM PROGRAM

	ENGINEERING TECHNOLOGY DIVISION (ETD)											
	PROGRAM MANAGER	EXPERIMENTAL ENGINEERING SECTION	PROJECT LINE† PERSONNEL	STRUCTURAL MECHANICS SECTION	QA COORDINATOR	GENERAL ENGINEERING COORDINATOR	FUSION ENERGY DIVISION	M&C DIVISION	I&C DIVISION	FIRST-WALL REVIEW COMMITTEE	SUPPORTING PARTICIPANT	GENERAL ENGINEERING DIVISION
PROGRAM MANAGEMENT	1*	4	4			3	4			4		
PROGRAM GUIDANCE	2	3	4			3	2	4		1		
TECHNICAL ASSURANCE	1	3	2	3	4	1	4	4	4	4		3
QUALITY ASSURANCE	3	2	3		1	4				4		3
COORDINATION WITH OTHER FUSION PROGRAMS	2		3			1	2	3		3		3
HEAT SOURCE DEVELOPMENT	3	3	2		3	3	4		4	4	1	4
VACUUM CHAMBER REACTIVATION	3	1	2			4	4					
ASSEMBLY OF WATER FACILITY	3	1	2			4			3	4		3
ADAPTATION OF HELIUM FACILITY	3	1	2			3			3	4		3
PREPARATION OF MATERIALS FOR TEST SPECIMENS WITH SIMULATED IRRADIATION DAMAGE	3	3	2			3		1		4		
TH/TM TESTING	3	3	1	3		3	4	4	3	4		
MODULE QUALIFICATION TESTING	3	2	1	2		3	4	4		4	4	
INSTRUMENTATION AND DATA ACQUISITION	3	3	2	3		3	4	4	1	4	4	3
DEVELOPMENT OF ANALYTICAL MODELS AND DATA ANALYSIS	3	3	1	2		2	4			4		

†UNSPECIFIED NUMBER OF PEOPLE
*DEGREE OF PARTICIPATION
1 PRIMARY RESPONSIBILITY
2 MAJOR INVOLVEMENT
3 SHOULD CONTRIBUTE
4 ADVISORY ROLE

C. Characteristics of Phase-I Evaluations

We believe the 1000 cm^2 (suggested order of first-wall test surface area) should be viewed as a minimum. It is perhaps sufficient to provide some basic thermal hydraulic data, but we think it insufficient for later testing to establish failure modes and failure frequencies. The intention of this proposal would be to seek to provide the capability of testing the maximum size test piece limited by our existing vacuum chamber (1.2 M diam. × 3.35 M high) and by the corresponding required radiant heat source. We believe this could be of the order of a factor of 5 larger than the 1000 cm^2.

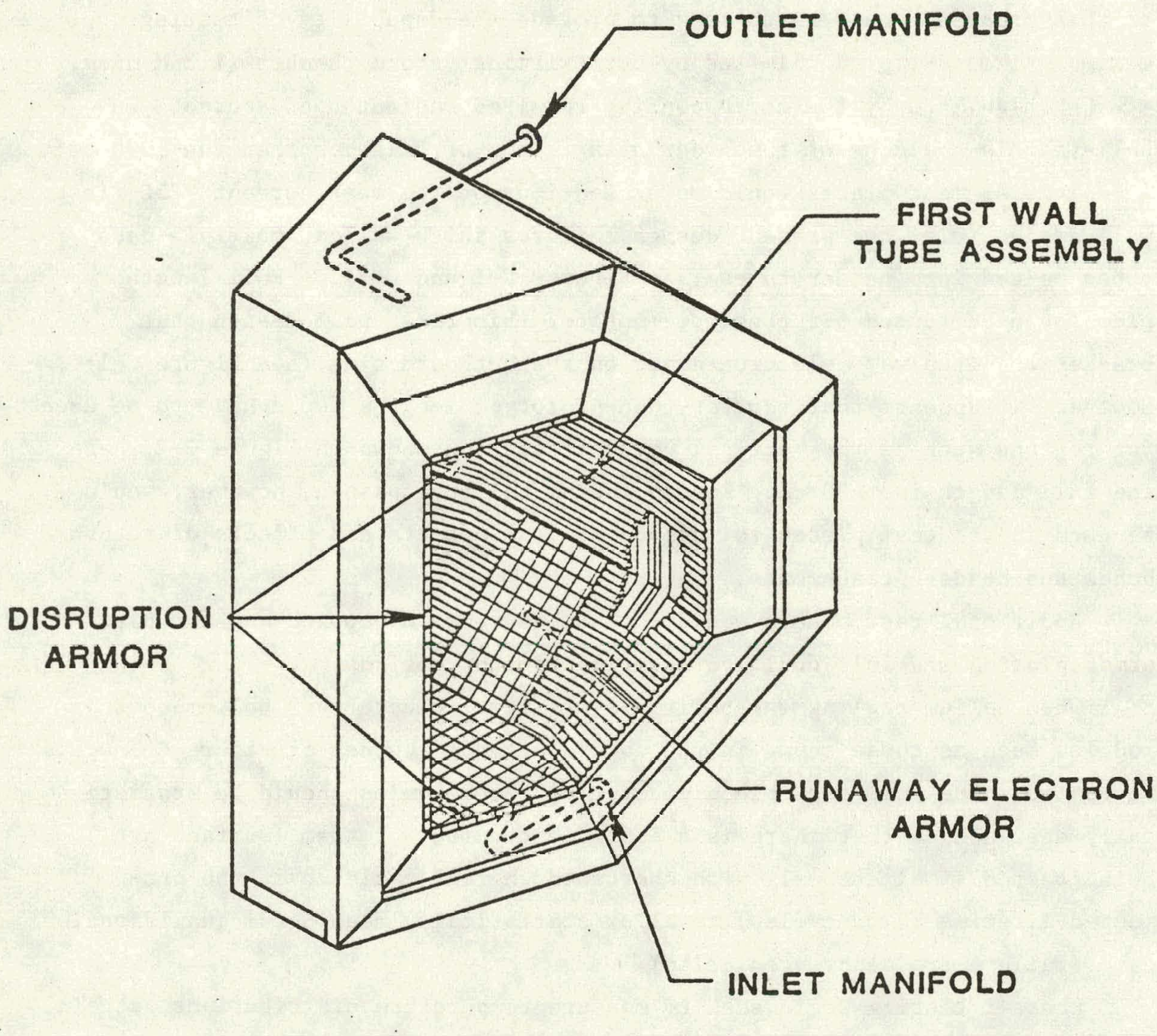
Initial test pieces would model sections of the most current FEDC first wall design (e.g. the present design features thick-walled, multiple-pass tubes welded into header tubes (see Figures C-1 and C-2). Full length pieces can be tested for those segments of this first wall design that bracket the runaway electron armor on the outboard side (see Figure C-1). However, it appears that shorter-than-prototype lengths would have to be used for regions such as the vertical outboard section shown on Figure C-2 where the full length is ∿420 cm. Enough individual tube passes, however, would be used in the test pieces to include all end effects and effects of return bends and header attachments.

Additional test pieces would include segments of cooled and uncooled armor plating and full scale versions of limiter designs.

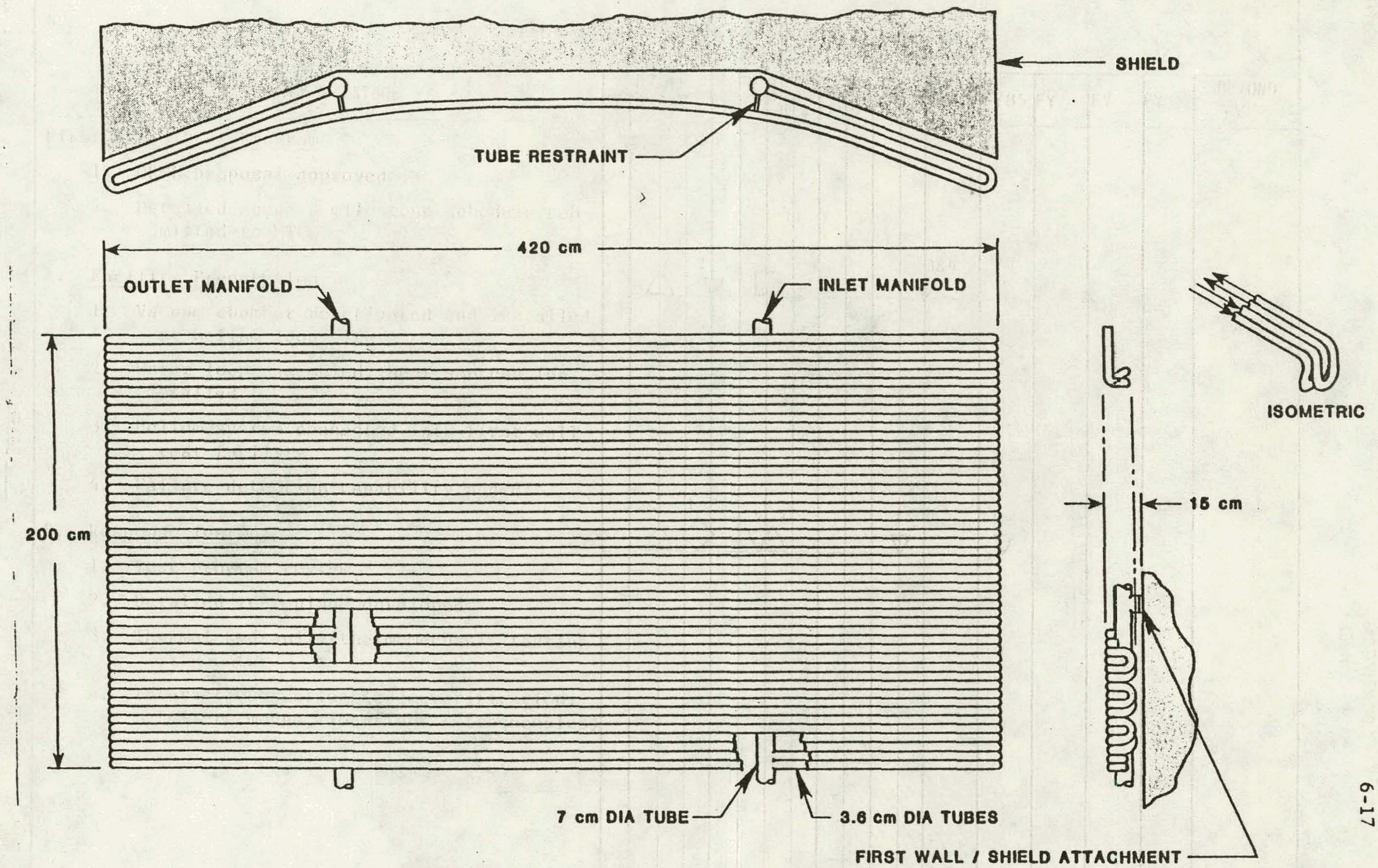
When helium cooling capability is utilized, clusters of helium-cooled modules, such as those shown in the lower left-hand corner of Figure C-3, would be tested. The full test piece cluster of such modules should be statistically designed so that there is a sufficient number of them (surface area) being tested simultaneously with shortened thermal cycles over the projected lifetime (>10^5 cycles) to allow statistically meaningful qualification data (failure frequency prediction).

Phase-I testing would seek to measure temperature distributions, strain distributions (displacements), and heat transfer coefficients under cyclic loading, assymetric heating conditions with independent variations of: the magnitude and frequency of heat flux cycling, test piece geometry (e.g. tube wall thickness, structural restraints, bends, entrance and exits into headers, etc.), and coolant flow conditions (e.g. Reynolds number and flow distributions).

Fig. C-1
FIRST WALL AND
ARMOR ARRANGEMENT



TYPICAL FIRST WALL TUBE PANEL



MATERIAL - 316 STAINLESS STEEL

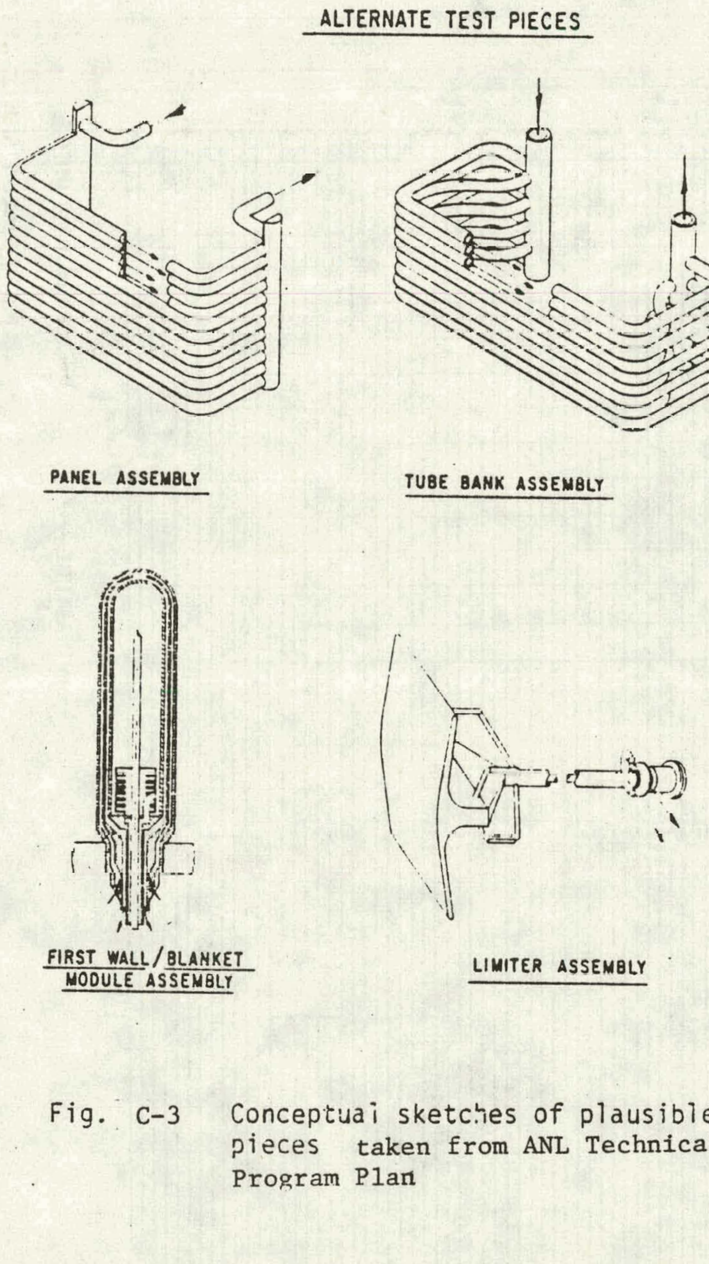


Fig. C-3 Conceptual sketches of plausible test pieces taken from ANL Technical Program Plan

Under appropriate vacuum conditions and with simulated irradiation-damaged test pieces, later testing would extend to failure qualification (failure modes, locations, and frequencies) as related to the cyclic loading conditions.

With a hot ion source, additional testing would involve effects of plasma disruptions, ion impingement onto entrance regions of divertors (and other divertor areas) and (possibly) assessment of divertor heat dump surface cooling.

Section H of this proposal presents additional information on the capabilities of the proposed facilities and projected testing conditions.

D. Supporting Participants

ORNL in general, and their fusion programs specifically, through many previous successful inter-relationships, has demonstrated the desire, capability, and willingness to work cooperatively with industry, universities, and other laboratories.

For the First Wall TPE-I proposal as outlined here, we believe the most benefitial involvement of supporting participants for Phase-I would be in providing the appropriate heat sources required in the program to reproduce the fusion environment conditions.

We are aware of similar requirements in particular aero-space industry and of their development and use of graphite element radiant heaters (with heat fluxes in excess of 200 W/cm^2 onto large surfaces) and high power arc heaters that might simulate plasma disruption heat source.

Regardless of whether the funding level were 200 K or 300 K in FY 81, the ORNL proposal would be to approach the appropriate aero-space industry as soon as program approval is obtained to determine the feasibility, cost, and schedule for them to supply the program with a radiant heat source.

E. Schedule

A proposed TPE-I schedule is presented in Table E-1. for five separate task areas: facility preparation, response testing, simulation model development, qualification testing, and material development for simulation of irradiation damage. This schedule is indicative of what we think is

Table E-1

NO.	189 SUBTASK/MILESTONE SCHEDULE																	

SUBTASK/MILESTONE	FY 81				FY 82				FY 83				FY 84	FY85	FY	FY	FY	BEYOND FY
	1	2	3	4	1	2	3	4	1	2	3	4						
First Wall TPE-I Program	▼	△ 1		▽ 2														
A. Facility Preparation		△		▽ 1		□ 2						▽ 3&4						
B. Response Testing		△		◇ 1		▽ 2	△ 3				▽ 3			▽ 4				

First Wall TPE-I Program

1. ORNL proposal approved
2. Detailed scope & milestone schedule submitted to MTCC

A. Facility Preparation

1. Vacuum chamber reactivated and installed at helium loop site
2. Water loop assembled, heat sources installed
3. Helium system connected into first-wall test facility
4. Failure detection capability added

B. Response Testing

1. Test program review
2. Detailed test plans developed
3. Thermal hydraulic/thermomechanic testing with water
4. Interactive testing (code verification; failure modes, locations & frequencies)

△: Begin Project □: Major Milestone

▽: lend Project

PAGE ___

TITLE:

ACTIVITY NO. 189 No.

Table E-1 (cont'd)

NO.

189 SUBTASK/MILESTONE SCHEDULE

SUBTASK/MILESTONE	FY 81				FY 82				FY 83				FY 84	FY85	FY 86	FY 87	FY88	BEYOND FY 88
	1	2	3	4	1	2	3	4	1	2	3	4						
C. Simulation Model Development		△				1▽		2▽		3▽								
1. Heat transfer & thermal stress codes combined																		
2. Parametric studies for alternate designs																		
3. Failure prediction added																		
4. Code/test interaction																		
D. Qualification Testing (To Failure)														1△	1▽ 2△	2▽		
1. FED prototype test																		
2. Second prototype test																		
E. Test Piece Preparation For Simulation Of Irradiation Damage Effects		△		1▽ 2△								2▽		3□	4▽			
1. Initial definition of irradiation damaged material properties																		
2. Develop simulated radiation damaged materials																		
3. First test module fabricated																		
4. Second test module																		

TITLE:

ACTIVITY NO. 189 No.

realistic but must be considered as preliminary at this time. As indicated in Table E-1, a firmer, more detailed, scope and milestone schedule will be developed and submitted to MTCC during the second half of FY 81.

F. Identification of Key Personnel

The following professionals have been selected to play key roles in conducting the TPE-I program should ORNL be selected as the subcontractor:

Role	Name (% time)
1. Program Manager	Dr. T. S. Kress (25% phased into 100% as program expands)
2. Assurance of Technical Significance to Magnetic Fusion Energy Program	T. J. Huxford (50% phased into 100%)
3. Responsibility for Design, Assembly, and Operation of Test Facilities	R. E. MacPherson (10%;- 100% from direct line staff member)

Biographical sketches of these key personnel are included as Appendix A. On Figure B-1 (Section B), a representative matrix organizational structure, the above three are associated respectively with the items identified as Program Manager, General Engineering Coordinator, and Experimental Engineering Section.

Because we view the establishment of a strong first-wall testing and qualification program to be of critical need and of high priority at this stage of the national fusion energy effort, other key personnel alluded to on Figure B-1, especially those identified as "project line personnel," will be selected for their demonstrated abilities and experience in areas of engineering experimentation, heat transfer, fluid mechanics, and structural mechanics relevant to fusion energy. The in-house "First-Wall Review Committee" would be selected from the many excellent and involved experts in the various ORNL fusion energy programs. They would be selected based on their position, knowledge, and potential contribution to the program. The following is a representative list of potential members for this committee in alphabetical order:

L. A. Berry (EBT-P)
E. H. Bryant (Engineering)
J. M. Corum (Structural Mechanics)
H. H. Haselton (Plasma Technology)
O. B. Morgan, Jr. (Fusion Energy Division)
R. Onega (Plasma Disruptions)
J. L. Scott (Metals and Ceramics Division)
J. Sheffield (Tokamaks - ISX-C)
D. E. Steiner (FEDC)
H. E. Trammell (Engineering Technology Division)

ORNL management has pledged its support to the first-wall program and has given assurance that these people (or others in the various ORNL fusion energy programs) may be approached to determine their availability to serve on such a review committee. Many have already expressed their willingness to serve in this capacity. Selection of and interaction with, a vital, relevant high-level internal committee at ORNL should not be a problem.

G. Technical and Managerial Capabilities of ORNL

The fusion work at ORNL, which supports nearly all phases of the national toroidal fusion program, is unique in its breadth. This broad effort in plasma physics experiment, theory, and technology development, conducted in the context of advanced design activities, gives ORNL a major role in the development of fusion energy.

Since 1969 the program has emphasized tokamaks and the supporting technologies. The neutral beam technology developed by the laboratory has contributed substantially to recent advancements in tokamak development.

Other contributions are being made through the development of large superconducting coils (the Large Coil Test Facility, LCTF), and design through the Fusion Engineering Design Center (FEDC). In addition the ELMO Bumpy Torus (EBT), an ORNL concept, has recently been designated the leading alternative to tokamaks and mirrors and has been selected by DOE for advancement to the proof-of-principle phase. ORNL will assume management responsibility for the proof-of-principle experiment, the EBT-P, which will be designed, constructed, and operated by industry.

The strength of ORNL's fusion program lies in its integration. By coordinating basic physics research with engineering and technology development, ORNL is uniquely able to address the problems of physics-technology interfaces - an area in which the first-wall clearly falls.

Although ORNL's fusion energy involvement may be well known, other capabilities may be less apparent. ORNL has had an active liquid metals development program supporting initially the Aircraft Nuclear Propulsion Program, then the Space Program, and now the Liquid Metal Fast Breeder Program. We have been continuously active in support of the helium-cooled High Temperature Gas-Cooled Reactor and the Gas-Cooled Fast Breeder Reactor. Thus, there is a familiarity with helium and liquid metal technology paralleling that of fusion technology. These activities have led to a wealth of expertise and experience in design and operation, handling and safety, materials behavior characteristics, heat transfer and fluid mechanics, instrumentation, components, and application of special techniques.

ORNL is operated under the Union Carbide Corporation management system ("management by objectives") and utilizes a matrix structure (project and line) to make optimum use of its interdisciplinary character. We have demonstrated our management capability and have had much success in involving and cooperating with industry and other institutions on a variety of sophisticated engineering missions.

H. Facilities and Resources for the First-Wall TPE-I Program

There exists considerable equipment and facilities at ORNL that would be applicable and can be committed to the First-Wall Program. A complete inventory and identification of these would constitute an early part of the TPE-I effort. The various ORNL divisions and the facilities of the ORNL fusion energy programs (e.g. the neutral beam development program and the ISX-C) will be investigated to determine if they have appropriate equipment resources that may be brought to bear on the First-Wall Program. At this time, ORNL has identified the following to be made available to the First-Wall Program:

H-1: Vacuum Chamber

The Engineering Technology Division (ETD) of ORNL has committed an existing vacuum facility to the First-Wall Program. This General Electric Vacuum Chamber was previously used to provide a space simulating environment in tests of refractory metal corrosion loops which operated at high temperature to simulate application in space power systems. It was shut down in 1970 and has been maintained in standby status since that time. The general specifications for the system are:

chamber size.	1.2 M (4 ft) diam. × 3.35 M (4 ft) high	
chamber pressure [empty	5×10^{-11} torr	
[with experiment	10^{-9} torr	
evacuation system	sublimation pump	20,000 ℓ/s
	ion pump	2,400 ℓ/s
	mechanical pump	25 ℓ/s

A copy of the original procurement specification is included in Appendix B.

For the First-Wall application, this chamber would be re-activated, all systems checked for proper operation, and the assembly moved to the proposed site for the First-Wall Program (co-location with helium loop). Test pieces, heat sources, and shutters (if required) would be designed to be compatible with and fit within the vacuum chamber.

H-2: Water Cooling:

A water circulating and conditioning system is required for Phase-I testing in support of FEDC. From our preliminary assessments, we believe the following capabilities are desirable for this program:

pumping capacity	<500 gpm
cooling capacity (water cooled heat exchanger) . .	1 MW
coolant inlet temperature	40°C

coolant outlet temperature		100°C
pre-conditioning water heater	}	capacity and sizes to be determined compatible with above
valves		
piping		
flowmeters		
pressure measurement		
temperature measurement		

As the result of many years of engineering testing, the ETD (and ORNL in general) has much existing water circulating and conditioning equipment. Such existing equipment will be made available to the First-Wall Program and an appropriate water loop will be assembled at the First-Wall site along with the vacuum chamber and the nearly complete helium loop (see below).

H-3. Helium Cooling:

Helium presently appears to be the front-running choice as primary coolant for a fusion reactor. The FEDC device, while water cooled for practical and operational reasons, will most probably have test segments devoted to testing helium cooling concepts. To pre-qualify these for FEDC and to meet the objective of providing the ultimate first-wall design and qualification data base, helium cooling capability is required for a complete first-wall program.

While water cooling loops can easily be established, the capability of testing with high pressure helium is more difficult and expensive to provide.

Fortunately, as part of the DOE sponsored Gas-Cooled Fast Reactor (GCFR) Program at ORNL, the Engineering Technology Division is presently completing the installation of an extensive helium circulating testing facility that has been designated as the Core Flow Test Loop (CFTL). The significant characteristics and a schematic flow diagram for the CFTL are presented as Appendix C.

The original mission of the CFTL was to provide the capability for high-temperature, high-pressure, steady-state and rapid transient testing of simulated fuel rod bundles for advanced gas-cooled fast breeder reactors. However, in FY 1980 as the CFTL approached completion (∿$9 1/2 M expended on loop construction), the GCFR support from DOE was reduced leaving enough funds to complete a modified version (still more than adequate for First-Wall). Additional support was obtained through the High Temperature Gas Cooled Reactor (HTGR) Program of DOE to investigate the possible utilization of the CFTL for HTGR application. Assuming the HTGR assessment culminates in a proposed HTGR usage, then the First-Wall utilization of this facility would involve sharing with the HTGR program. The sharing mode would have to be established but is anticipated to require the addition of by-pass piping and control valves that would provide helium from the main CFTL circuit to be fed into the First-Wall Test pieces and to provide the appropriate instrumentation interfaces for sharing the Data Acquisition System (see below).

H-4. Data Acquisition:

A viable First-Wall testing facility for TPE-I will need the capability for making a variety of measurements under steady-state and transient conditions. These should include:

- coolant flow rates, temperatures, and pressures;
- temperature distributions in the first-wall structures and in the coolant volumes;
- strain distributions and displacements in the first wall structure;
- heat source powers and temperatures;
- chamber vacuum, and (eventually)
- crack-initiation, growth, and failure detection.

It would not be a trivial matter to start from scratch and provide the techniques, instrumentation, signal conditioning, and data acquisition systems for the above. In addition to its helium circulating equipment, the CFTL includes a high speed (10 Khz) 640 channel, PDP-11 computer controlled data acquisition system that would be utilized in the First-Wall Program as shared with the HTGR Program. ORNL has a strong Instrumentation and Controls Division that can provide the required expertise and general support to the First-Wall Program.

I. Data Restrictions

All data and results of the First-Wall TPE-I Program will be released on an unrestricted basis to the government. We do not anticipate any requests for proprietary or other restricted status to any parts of the program.

J. Labor and Overhead Rates

ORNL is a government owned facility operated by Union Carbide Corporation - Nuclear Division. As such, the rate structures are evaluated and approved by DOE. For FY 81 planning purposes, the hourly rates for various service and craft divisions of ORNL are shown in Table J-I. The actual overhead rates that occurred during FY 80 are shown in Table J-II. A typical budgetary estimate based on a 24 man/month level of effort (split equally between technical and craft labor) would appear as shown in Table J-III.

Table J-III is a hypothetical budget to illustrate a simplified typical cost breakdown for a 1 man-year technical and 1 man-year craft labor level-of effort project. It should be used for comparative purposes and should not be viewed as an actual budget estimate for the First Wall TPE-I Program. Table J-III does not include any subcontracting to supporting participants. In the ORNL proposal, the significant supporting participant would supply the radiant heat source required for the First-Wall TPE-I testing. The cost and schedule for this item would be subject to the contract negotiations between ORNL and the supporting participant.

ORNL cannot enter into a cost sharing arrangement. However, it should be recognized that significant facilities and equipment will be made available to the program at no additional cost except for any required modifications, maintenance, or operating expenses.

Table J-I

ORNL
STANDARD SERVICE RATES FY 1981

Instrumentation and Controls, Quality Assurance and Inspection, Plant and Equipment and Analytical Chemistry Services

	I&C		QA & I	P&E			ORNL	ANALYTICAL
Expense Funding	ENGR.	MAINT.		RSCH.	MAINT.	APPRENTICE	ENG.	CHEMISTRY
Base Rate	$30.85	$22.50	$25.00	$23.90	$19.55	*	*	$32.80
G&A/GPS (31%)**	9.56	6.98	7.75	7.41	6.06			10.17
Total Expense Rate Per Hour	$40.41	$29.48	$32.75	$31.31	$25.61			$42.97
Capital Funding								ORNL Research Divisions
Base Rate	$14.77	$11.48	$12.20	$10.15	$ 9.45	$ 7.80	$12.20	$17.00
Incremental Overhead (98%)	14.47	11.25	11.96	9.95	9.26	7.64	11.96	16.66
Non-Productive Time	.74	.57	.73	.61	.57	.47	.52	—
Total Capital Rate Per Hour	$29.98	$23.30	$24.89	$20.71	$19.28	$15.91	$24.68	$33.66

* Actual monthly rate applied.

** The G&A/GPS rate indicated is a tentative planning rate which is subject to change.

Table J-II

FY 1980 OVERHEAD RATES

Category	Oct	Nov	Dec	Jan	Feb	Mar	Apr	May	Jun	Jul	Aug	Sep	FY 1980 Average	
[a]Company Plans and Benefits	25.1	23.5	19.1	26.3	26.8	26.7	26.5	25.9	24.5	23.4	25.5	24.7	24.8	
[a]Division Administration	28.7	24.2	20.7	29.4	26.7	24.3	24.6	22.7	28.8	33.9	33.9	29.1	27.2	
[b]General and Administrative	10.9	13.4	12.6	13.4	12.9	12.7	13.7	12.2	11.3	12.7	14.0	15.7	13.0	30.2
[b]General Plant Services	18.3	15.5	15.9	16.1	16.0	17.5	16.6	16.7	17.0	18.5	17.7	20.9	17.2	

[a]Applied at Cost Center level on net labor.

[b]Applied at R&D Account level on all elements of cost except Major Material-Exempt and R&D Subcontract-Exempt.

xc. JLW

Table J-III

Illustrative Cost Breakdown for a Hypothetical
1 Man-Year Technical and 1 Man-Year
Craft Level-of-Effort

Item		(MY)		Cost($1000)
Direct Manpower	[Technical	1		60
	[Craft	1		45
Materials (loop assembly and test piece construction)				50
Technical Publications, computers, travel, misc.				5
			Subtotal	160
Overhead (General and Administrative; General Plant Services)				50
			Total	210

APPENDIX A

BIOGRAPHICAL SKETCHS OF KEY PERSONNEL FOR THE FIRST-WALL TPE-I PROPOSAL

Name: Dr. Thomas S. Kress
Manager, Aerosol Release and Transport Program
Group Leader, Advanced Concepts Development Section
Engineering Technology Division
Oak Ridge National Laboratory

Education: Berea College, Berea, KY, 1951-1952 (no degree)
University of Tennessee, Knoxville, TN, B.S. Mech.Engr. 1956
University of Tennessee, Knoxville, TN (including graduate study at Rensselaer Polytechnic Institute - Hartford Graduate Center), M.S. Mech.Engr. 1965
University of Tennessee, Knoxville, TN, Ph.D. Engr. Science, 1971
Plus: Union Carbide Corporation Management (MBO) Training Course and additional technical and managerial "short courses"

Experience: 1974 to present. Program Manager - Aerosol Release and Transport Program;
Group Leader - Advanced Concepts Development Section, Engineering Technology Division

1) Fuel and fission product release under core-melt accident conditions
2) Molten fuel-liquid sodium coolant interactions
3) Fuel aerosol, fission products, and sodium combustion aerosol behavior in primary and secondary containments

1972 to 1974. Heat transfer/fluid mechanics specialist - Engineering Technology Division

1) Sodium boiling, heat transfer, and thermal hydraulics in LMFBR cores
2) Effectiveness of emergency core cooling systems for PWRs

1968 to 1972. Engineering Specialist - Reactor Division - on Molten-Salt Reactors Programs;

1) Mass transfer from liquids to dispersed bubbles (fission product stripping from molten salt fuel)
2) Generation and removal of small gaseous bubbles in molten salts

1965 to 1968.

1) Gas-cooled reactors heat transfer and safety
2) Steam oxidation of reactor structural graphite

1963 to 1965.

1) Experimental and analytical studies of two-phase flow under boiling conditions
2) Experimental and analytical studies of fission product transport and deposition

1959 to 1963. ORNL - Reactor Division

1) In-pile experiment design, construction, testing, and hazards evaluation
2) Heat exchanger design, construction, and testing
3) Digital and analog computer programming

Experience (cont'd): 1956 to 1959. Pratt and Whitney Aircraft Company.

1) Optimization parametric studies for reactor fuel elements
2) Heat transfer and fluid dynamics testing and analysis related to Aircraft Nuclear Propulsion core designs
3) Advanced reactor propulsion designs for missils and rockets

Professional Activities:

1) American Society of Mechanical Engineers (Member, past section chairman, and past regional committee chairman)
2) American Nuclear Society
3) National Society of Professional Engineers (registered P.E. in state of Tennessee)
4) American Association for the Advancement of Science
5) Chairman - Aerosol Properties Committee of the SACRD Data Base Program
6) Member of CSNI Specialists Group on Nuclear Aerosols in Reactor Safety

Patent: 1974. "An Electric-Type Heater for Producing a Given Non-Uniform Axial Power Distribution." This patent is for an electric cartridge type heater that simulates a reactor fuel pin.

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1972 "Mass Transfer Between Small Bubbles and Liquids in Cocurrent Turbulent Pipeline Flow," ORNL-TM-3718 (1972).

1973 "Effects of Partial Blockages in Simulated LMFBR Fuel Assemblies," (with M. H. Fontana, L. F. Parsly, D. G. Thomas, and J. L. Wantland), ORNL-TM-4324, December (1973).

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1975 "Temperature Distribution in a 19-Rod Simulated LMFBR Fuel Assembly in a Scalloped Duct (Fuel Failure Mockup Bundle 1A) — Record of Experimental Data," (with M. H. Fontana, P. A. Gnadt, R. E. MacPherson, L. F. Parsly, and J. L. Wantland), ORNL-TM-4670, (1975).

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1975 "Temperature Distribution in a 19-Rod Simulated LMFBR Fuel Assembly in a Scalloped Duct (Fuel Failure Mockup Bundle 1B) — Record of Experimental Data," (with M. H. Fontana, R. E. MacPherson, P. A. Gnadt, L. F. Parsly, and J. L. Wantland), ORNL-TM-4939, November (1975).

1976 "Temperature Distribution in 19-Rod Simulated LMFBR Fuel Assemblies with and without an Edge Blockage (Out-of-Reactor Tests for ANL SLSF P1 Experiment) — Record of Experimental Data for Fuel Failure Mockup Bundles 5B and 5C," (with M. H. Fontana, P. A. Gnadt, R. E. MacPherson, L. F. Parsly, and J. L. Wantland), ORNL-TM-5003, March (1976).

1976 "Temperature Distribution in a 19-Rod Simualted LMFBR Fuel Assembly with a Six-Channel Internal Blockage (Fuel Failure Mockup Bundle 3A) — Record of Experimental Data," (with M. H. Fontana, P. A. Gnadt, R. E. MacPherson, L. F. Parsly, and J. L. Wantland), ORNL-TM-5101, March (1976).

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Name: Robert E. MacPherson, Jr.
Head, Experimental Engineering Section
Engineering Technology Division, Oak Ridge National Laboratory

Universities Attended:
Allegheny College, Meadville, Pa B.S. Chem 1943
University of Pittsburgh, Pittsburgh, Pa B.S. Ch. E. 1948
Oak Ridge School of Reactor Technology - 1958-59

Experience:
1951 to present. Chemical Engineer, Oak Ridge National Laboratory, Oak Ridge, Tennessee
1) development of components such as heat exchangers, radiators, pumps, valves, instruments and controls for high temperature gas, molten salt, and liquid metal reactor systems, 1951 to present
2) supervision of chemical engineering tasks leading to better understanding of compatibility problems between coolants, graphites, and structural materials in gas-cooled reactor systems, 1957 to present
3) supervision of operation of one- and two-phase liquid metal corrosion tests, 1960 to present
4) supervision of development of systems for specialized reactor power generation concepts, 1962 to present
5) assistant manager of an engineering group having approximately 30 technical members, 1962 to 1965
6) Head, Experimental Engineering Section, Engineering Technology Division, 1965 to present

1948-1951. Chemical Engineer, Standard Oil Company (no Exxon)
Linden, New Jersey
1) pilot plant data correlation and reporting
2) supervision of fluid catalytic cracking pilot plant operation, 1950-1951

1943-1947. Chemist, Tennessee Eastman Corporation, Oak Ridge, Tennessee
1) uranium recycle process development, 1943-1945
2) shift supervision of uranium recovery and reprocessing plant, Manhattan Project 1945-1947

Professional Organization: American Nuclear Society

Professional and Academic Honors:
1976 Tenn. Soc. Prof. Engr.-Significant Achievement Award
1943 Lee Scholarship - Chemistry
1939 Prize Scholarship - Allegheny College

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PATENTS

PATENTS GRANTED

1953 CONTINUOUS FLUID HYDROFORMING
1953 R. E. MACPHERSON, JR. PAT. NO. [illegible]
1953 USA

TOTAL PATENTS GRANTED 1

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1974 IN A SODIUM-COOLED 19 ROD BUNDLE NWITH J. L. WANTLAND, M.
1974 H. FONTANA, P. A. GNADT, T. S. KRESS, AND L. F. PARSLY),
1974 TRANS. AMER. NUCL. SOC. 19, 245 (1974)

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Oklahoma State University, Mech. & Aerospace Engr. B.S. 1972
Oklahoma State University, Mech. & Aerospace Engr. M.S. 1974

Experience: 1974 to present. Mechanical Design Engineer, specializing in the thermal hydraulic design of both conceptual and experimental fusion related heat transfer equipment.

1971 to 1972. Mech.Engr.- C.H. Guerwsey Consulting Engineers. Performed long-range planning studies for municipal electric utilities.

1966 to 1969. Development technician, Heat Transfer Laboratory, Air Research Mfg. Co. Designed, set-up and operated test stand facilities where a variety of heat exchanger R&D was performed for aerospace applications.

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APPENDIX B

PROCUREMENT SPECIFICATIONS FOR ULTRA-HIGH VACUUM SYSTEM

I. TITLE

G. E. Vacuum Facility

II. LOCATION

9201-3, 2nd floor, Track Floor Area, N. E. corner.

III. RESPONSIBLE PERSON

W. R. Huntley

IV. DESCRIPTION

The General Electric vacuum chamber was purchased in 1965 to provide chamber pressures similar to the vacuum in space to prevent oxidation of the outer surfaces of refractory metal corrosion loops which operated at very high temperature to simulate applications in space power systems.

V. SPECIFICATIONS

Chamber size	- 4 ft diam by 11 ft high	
Chamber pressure empty	- 5×10^{-11} torr	
With experiment	- 1×10^{-9} torr	
Evacuation System	- Sublimation pump	20,000 ℓ/s
	- Ion pump	2,400 ℓ/s
	- Mechanical pump	25 ℓ/s
Reference Drawings	- I-10436-QG-004E-1	
	I-10436-QG-001E-3	

VI. STATUS

The system was shutdown in 1970. The main flange is not tightly bolted, and an argon purge is maintained into the chamber to retard reaction of a thin layer of lithium residue which is present on the walls of the chamber.

PROCUREMENT SPECIFICATION

FOR

ULTRA-HIGH VACUUM SYSTEM

1.0 SCOPE

1.1 This specification is for an Ultra-High Vacuum System to be used for long-term test of refractory metal tubing carrying molten alkali metal.

1.2 The system shall have the following general characteristics:

A. CLEAN -- System must be completely oil free in the final roughing and ultra high vacuum phases of operation.

B. BAKEABLE -- System must be completely bakeable to a temperature of 250°C.

C. ADAPTABLE -- System must adapt to a wide variety of experimental parameters.

D. VIBRATION -- System is to be vibration free in the high vacuum phase of operation.

E. RELIABILITY -- System must be capable of maintaining high vacuum for long periods of time and withstand emergencies such as power failures. The vacuum pumping section of the system must be unaffected by sudden exposure to atmospheric pressure.

F. PERFORMANCE -- System shall attain an ultimate pressure of at least 5×10^{-11} Torr in a period of 24 hours.

1.3 Seller shall furnish evidence that he has competence in system design of the type delineated by this specification and submit a proposal that clearly explains his approach to this overall requirement.

1.4 The seller is to design, develop, deliver and conduct acceptance tests on an Environmental Research Chamber within the framework of this specification at the company's Oak Ridge, Tennessee facility.

2.0 DESIGN DESCRIPTION

2.1 The system and all components contiguous with the high vacuum environment shall be fabricated of type 304 stainless steel with all junctures either inert gas welds or O.F.H.C. copper gasketed flanges and constructed in accordance with the best procedures consistent with the state of advanced vacuum art.

2.2 With the exception of the main chamber seals, the flanges are to be of the conflat type configuration, sexless, completely interchangeable within standard size groups and compatible with a wide variety of commercially available ultra-high vacuum feedthroughs.

2.3 The main system seals shall be of the captured gasket type such as the Varian Wheeler flange. The sized O.F.H.C. copper gasket is to be precisely positioned in the same place of the flange configuration for each closure and the flange pair must be made ~~in such a way as to obviate~~ mechanical warping or coning. The seller shall be prepared to explain the design of any other seal configuration that varies from the Wheeler type and show that they have been built in the size required and are currently in satisfactory use.

2.4 The overall vacuum system shall be twelve feet tall exclusive of supporting structures and four feet in diameter. It shall be nominally divided into three equal parts as follows:

A. Upper environmental chamber
B. Central loop feedthrough chamber
C. Lower pumping chamber

3.0 DESCRIPTION OF UPPER ENVIRONMENTAL CHAMBER

3.1 Chamber shall be nominally four feet tall and four feet in diameter and constructed as described in Article 2.1. It shall terminate at the top in a lifting attachment assembly capable of supporting 150% of the total system weight.

3.2 At the bottom it shall terminate in a flange such as described in Article 2.2 for mating with the top flange of the central chamber.

3.3 The outer walls shall have an array of electrical bake-out heaters for evenly heating the holding the chamber to 250°C and water cooling channels. The latter are to be capable of absorbing and dissapating at least 10 KW of heat per hour and must be fabricated such that the water is separated from the vacuum environment by two layers of metal. more

INSULATED AND

3.4 Externally the chamber shall be surrounded with a well-crafted polished aluminum heat shroud.

4.0 DESCRIPTION OF CENTRAL CHAMBER

4.1 The central chamber shall be nominally four feet tall, four feet in diameter and constructed as described in Article 2.1. It shall terminate at the top in a flange as described in Article 2.2. suitable for mating with the upper chamber and at the bottom in a flange as described in Article 2.2 for mating with the flange of the lower pumping chamber.

4.2 The outer walls shall have an array of electrical heaters for evenly heating and holding the chamber to 250°C and water cooling channels. The latter are to be capable of absorbing and dissapating at least 10 KW of heat per hour and must be fabricated such that the water is separated from the vacuum environment by two layers of metal.

4.3 Externally the central chamber will be surrounded by a removeable heat blanket which shall serve both as a protective device and thermal insulation.

4.4 The central chamber shall have 51 access ports equally spaced around the periphery as delineated in the Union Carbide preliminary drawing entitled "Vacuum Vessel Spoolpiece Nozzle Arrangement".

A. The following access ports shall be capped off as per the Company's Drawing No.

B. The following access ports shall be fitted with 1½" I.D. flanges of the conflat configuration.

C. The following access ports shall be fitted with 6" I.D. flanges of the conflat configuration.

D. The following access ports will be fitted with ultra high vacuum feed-throughs:

PORT	FEEDTHROUGH	VARIAN MODEL NO.
4A	20-Pin Inst. Elect.	954-5013
4C	8-Pin Inst. Elect.	954-5012
4D	8-Pin Inst. Elect.	954-5012
6B	8-Pin Thermocouple	954-5015
14C	8-Pin Thermocouple	954-5015

ETC.

E. The following access ports will be fitted with special ultra high vacuum feedthroughs in accordance with the included company drawings:

PORT	FEEDTHROUGH	UNION CARBIDE DRAWING NO.
5B	High Current	1234 A
9B	High Current	1234 A
15D	High Current	1234 A

4.5 All flange welds shall be done in accordance with the procedures prescribed in Varian Vacuum Division Publication VAC 2099B or Vacuum Technology Section of Research and Development Magazine, Issue September, 1966.

5.0 DESCRIPTION OF LOWER PUMPING CHAMBER

5.1 The lower pumping chamber shall be a nominal four feet tall, four feet in diameter and constructed as described in Article 2.1. It shall terminate at the top in a flange as described in Article 2.2 suitable for mating with the lower flange of the central chamber.

5.2 It shall be elevated above the floor level approximately two feet in order to permit access to the under side and the elevating structure must be capable of supporting 150% of the total system weight.

5.3 The outer walls shall have a suitable array of electrical heaters for heating and holding the chamber to 250°C and water cooling channels. The latter are to be capable of absorbing and dissapating at least 10 KW per hour of heat and must be fabricated such that the water is separated from the vacuum environment by two layers of metal.

5.4 Externally the lower pumping chamber will be surrounded by a removeable, well crafted and polished aluminum heat shroud.

5.5 The lower chamber shall have various conflat flanged feedthrough ports penetrating the peripheral vacuum wall with feedthroughs installed as follows:

A. (1) 2½" port terminating in a 2½" bakeable roughing valve like Varian No. 951-5032.
B. (3) High voltage feeds like Varian No. 915-5020.
C. (2) Liquid Nitrogen feeds like Varian No. 954-5023.

5.6 The bottom of the lower chamber vacuum wall will have (10) 1½" I.D. conflat flanged access ports mounted close to the center axis as follows:

A. (4) Blanked off with mating flange
B. (6) Fitted with tri-filament titanium sublimation cartridges like Varian No. 916-5017.

5.7 Internally the system shall have projecting upward from the bottom at least six tri-filament titanium sublimation cartridges capable of delivering a total of not less than sixteen grams of Titanium. The cartridges will be mounted on O.F.H.C. copper gasketed flanges of the conflat type for mating with the flanged access ports described in Article 5.6.

5.8 The peripheral wall shall have built into it a sufficient number of diode type sputter ion pumping elements to yield a pumping speed of 3000 liters per second. The magnets required for the operation of the pumping modules shall be mounted externally so as to facilitate rapid cool-down after baking and minimize surface area in the vacuum environment.

5.9 Internally, the lower chamber will have a stainless steel cylindrical removeable liquid Nitrogen cryoshroud mounted coaxially with the vacuum wall. The shroud shall have an integral optical baffle and an effective area exposed to the titanium sublimation cartridges to yield a pumping speed of at least 35,000 liters per second in the high vacuum range at room temperature. The shroud shall have sufficient cryosurface to yield a pumping speed in the high vacuum range for water vapor in excess of 300,000 liters per second. Ease of removing and installing the shroud will be a design consideration.

6.0 ROUGH PUMPING

6.1 Reduction of pressure from atmospheric to approximately a millimeter is to be accomplished by a company owned suitably trapped mechanical pump.

6.2 Reduction from the low millimeter region to the low micron region is to be accomplished by cryo-sorption pumping. The sorption pump system is to be capable of reducing the pressure to a few microns in 30 minutes or less.

6.3 The sorption pump array is to be portable, self-contained, be independently valved, have a pressure measurement device, have its own dewar flask assembly and bake-out apparatus.

7.0 INSTRUMENTATION

7.1 Protective devices are to be included to provide fail-safe operation of the vacuum system. An overpressure relay is required that will cut off power to both the pumps, the bake-out heaters and the test loop heaters.

7.2 Pressure in the central chamber area will be measured in the region of 1 mm (Torr) down to 1×10^{-5}Torr by a high pressure ionization gauge and in the region of 1×10^{-3} to 2×10^{-11} by a low pressure Bayard-Alpert ionization gauge type. Both of these gauges are to be of the nude design and project directly into the vacuum environment.

7.3 Gas species will be identified qualitatively and quantitatively by a partial pressure gauge which can be operated through a mass range of at least 1 to 70 at ambient or while being baked at 250°C. The instrument must be capable of measuring both partial pressures and total pressures down to 2×10^{-11}Torr and serve as a leak detector.

7.4 The controls and instrumentation associated with the vacuum system will be mounted in a control console to be provided by the vendor. The controls to be furnished are as follows:

A. Ion Pump Controls
B. Titanium Sublimation Pump Controls
C. Ionization Gauge Controls
D. Partial Pressure Gauge Control
E. Bake-out Heater Controls
F. Main Power Control

7.5 The electrical service required for operating this system shall be 208 VAC, three-phase - 60 cycle. Wiring for power from the laboratory power source to the junction box of the system will be accomplished on the premises by the company electricians.

8.0 TESTING

8.1 The vendor is to supply a test protocol with his proposal which specifies the tests to be run both in his factory and in situ at the Oak Ridge facility.

9.0 SPARE PARTS AND INSTRUCTIONS

9.1 The vendor shall supply with his proposal a list of spare parts, specialized tooling and instructions he will include with the system.

10.0 In addition, the vendor shall supply with his proposal the following:

A. Price and Delivery
B. Statement of warrantee policy
C. Engineering drawings of vacuum system
D. Statement of experience with sorption pumps
E. Statement of experience with large getter-ion pumps
F. Statement of experience with Titanium Sublimation pumping
G. Statement of experience with large chamber fabrication
H. Statement of experience with large flange seals
I. Statement of size and reliability of high current power feedthroughs
J. Any other information which will aid in the evaluation of the proposal.

APPENDIX C

INFORMATION ON THE CORE FLOW TEST LOOP (CFTL) — AN EXTENSIVE HELIUM TESTING FACILITY NEARING COMPLETION AT ORNL

~~xc: MHF~~
Kress

OAK RIDGE NATIONAL LABORATORY

OPERATED BY
UNION CARBIDE CORPORATION
NUCLEAR DIVISION

POST OFFICE BOX Y
OAK RIDGE, TENNESSEE 37830

July 3, 1980

Dr. Frank E. Coffman, Director
Division of Development and
Technology
Office of Fusion Energy
Department of Energy
Washington, D.C. 20545

Dear Frank:

Per our phone conversation we are sending you an information package about the high-temperature, high-pressure, helium loop Core Flow Test Loop (CFTL) at Oak Ridge. At your convenience, we will make a presentation to you, your staff, and anyone interested. Please inform us of your preferred date for the presentation.

The information package includes general loop data, flow schematic and isometrics of the loop. Several photographs show the progress and investment made in the loop. Some of the major components are also shown: the gas bearing circulator, largest known of its kind, at a development cost of 2 M$; a schematic of the vortex shedding flow meter tested at ORNL to an extreme range of more than 100:1 and excellent accuracy; the large heat exchanger designed and manufactured here. These items show that the loop is backed with adequate capability in design, manufacture, base technology, cooperation with industry as the need may arise. Further, the loop is one of a series of water and sodium loops of similar magnitude. The experience and expertise is available and is being applied to the CFTL.

The status of the loop is that most major items are either in place or expected for delivery on schedule. The Stage I Loop is scheduled for beginning of shakedown in July 1981. The cost estimate for Stage I is 13.5 M$; however, this includes several items which are specific to the GCFR application only and a contingency in excess of 1 M$ which has not been tapped. At the end of FY-80 the total investment and commitments in the CFTL loop construction will amount to about 9.5 M$. In addition, about 7 M$ were utilized for development, analysis, and test planning and design associated with the loop.

If there is any further information or material we can provide please do not hesitate to call me or Uri Gat, the Program Manager (FTS 624-0559).

Very truly yours,

Uri Gat for

Paul R. Kasten, Director
Gas Cooled Reactor Programs

cc: Uri Gat w/attachment
A. G. Grindell
H. E. Trammell
D. B. Trauger

CORE FLOW TEST LOOP

CFTL

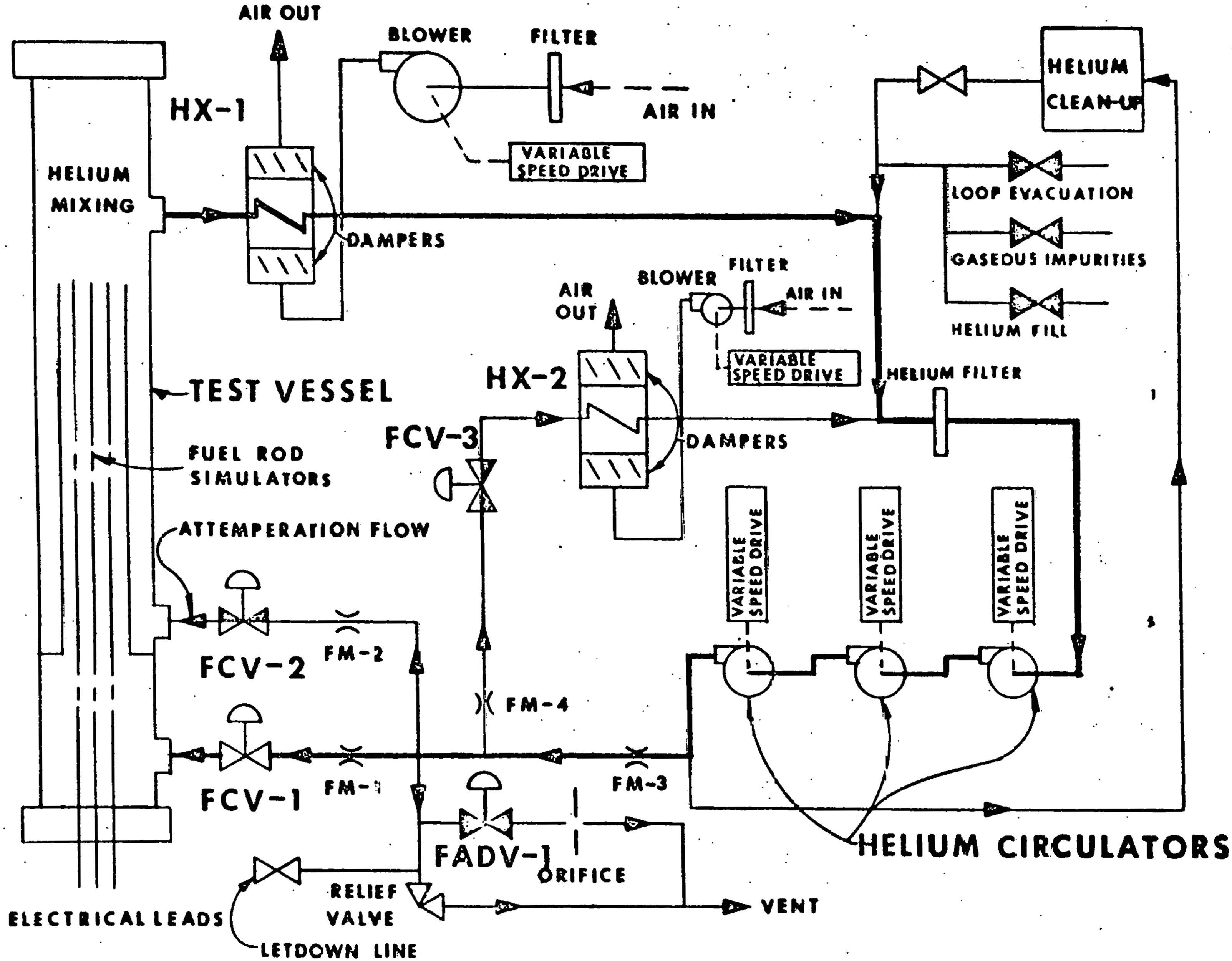
SIGNIFICANT CHARACTERISTICS

The Core Flow Test Loop — CFTL — is a high pressure, high power, high temperature, steady state and fast transients gas loop equipped with an extensive and sophisticated data acquisition system.

The main characteristics of the CFTL:

Pressure:	Ambient to 10.5 MPa (1550 psi)	
Power:	0 to 4 MW, controlled by 13 independent power supplies	
Temperature:	260 to 600°C with attemperation to stainless steel melting (approximately 1350°C)	
Transients:	Full power to zero power in	1 second
	Full flow to zero flow in	1 second
	Full pressure to approximately ambient less than	1 minute
	All transients fully program controlled	
Working Media:	Designed for helium with impurity control	
Flow Rate:	0 to 3.2 kg/s	
Data Acquisition:	High speed (10 kHz) 640 channels (expandable) computer controlled	
Location:	Building 9201-3, Y-12 Plant, Oak Ridge, Tennessee	
Operated by:	Engineering Technology Division, ORNL	
Sponsor:	DOE, GCFR Program	
Available:	Second half 1981 (partial capability). End of 1983 (full capability).	

Further Information: Uri Gat, ORNL (615) 574-0559; FTS 624-0559



FLOW DIAGRAM - CORE FLOW TEST LOOP

A-SK-RAM-5338 9-5-79

APPENDIX L

APPENDIX L

ARGONNE NATIONAL LABORATORY RESPONSE

U.S. Department of Energy
ARGONNE NATIONAL LABORATORY
9700 South Cass Avenue, Argonne, Illinois 60439 Telephone 312/972-5317

November 16, 1981
Ref: TPE II-25

Mr. Glen Longhurst
EG&G Idaho, Inc.
Idaho National Engineering Laboratory
P.O. Box 1625
Idaho Falls, Idaho 83401

Dear Glen:

In response to your questionnaire, the following information concerning ANL coolant loop facilities from which you can extract pertinent information is enclosed:

"Components Technology" Facilities

"Argonne Facilities - LMFBR Program"

"Building 206 Loop"

"High Temperature LMMHD Facility"

"Ambient Temperature LMMHD Facility"

The enclosed paper entitled "Chemical Processing of Liquid Lithium Fusion Reactor Blankets" give information on the lithium loop.

Yours truly,

H. Herman
Deputy Manager
First Wall/Blanket/Shield
Program

HH:mlw
Enclosure

cc: R. Zeno w/o Attch.
E. Sowa " "
M. Petrick " "
K. Schultz " "

bcc: C. Baker w/o Attch.
M. Abdou " "
R. Nygren " "
V. Maroni " "
D. Smith " "

Building 206 Loop

1.

a) The heat source is an electrically heated tungsten filament sandwiched between boron nitride plates.

b) The test environment is a chamber flooded with argon gas.

c) Approximate size accomodation volume is 2 x 2 x 1 ft.

d) No limitation on materials.

e) Temperatures can go as high as the melting temperature of tungsten, i.e., 3410°C. The maximum power level available is 300 kW.

f) The rise and decay time for the power is dependent upon the rapidity of electrical switching. The temperature rate of change depends upon the heat capacity and thermal heat transfer of the test piece.

g) Active cooling capability is approximately 400 kW with rejection to water.

h) Instrumentation includes strip chart, data acquisition systems and a H.P. 9845B Computer. Test sensors include thermocouples, pressure transducers, current, voltage, etc.

i) Currently in application to the PAHR program. Current costs involve primarily the attendant personnel, usually two persons.

j) Supporting facilities include ANL shops and analytical labs.

k) As assigned.

2a Closed loop configuration using Nak as heat transfer fluid with final heat rejection to water.

b) See above.

c) See above.

d) Flowrate maximum approximately 8 ℓ/min. Maximum stall pressure for the electromagnetic pump, 2 atm.

3. No nuclear heating available.

HIGH TEMPERATURE LIQUID-METAL MHD FACILITY (HT-LMMHD)

LOCATION: ANL-E, Argonne, Illinois

STATUS: Operational

PRINCIPAL USES:

The High-Temperature Liquid-Metal MHD Facility is used ~~to evaluate~~ sodium-nitrogen two-phase liquid-metal MHD generator performance as a function of temperature and other parameters.

FACILITY DESCRIPTION:

The facility provides known flows of liquid (sodium) and gas (nitrogen) for testing the two-phase liquid-metal MHD generator. It consists of a sodium loop and a nitrogen loop with a common mixer, test section, and separator/sodium dump tank. The sodium is recirculated, while the nitrogen is exhausted to the atmosphere from the separator through a scrubber to remove all traces of sodium. The nitrogen exhaust line is equipped with a control valve to fix the generator exit pressure. The sodium loop includes an electromagnetic pump, a flowmeter, control valves, and pressure transducers. The nitrogen loop comprises a heater and control system to heat the gas to the liquid temperature, a flowmeter, a flow-rate controller, and pressure transducers. Additional nitrogen is supplied for accessory cooling, and a dc iron-cored electromagent provides the magnetic field.

TEST CAPABILITY:

The test section fits between Graylok flanges located about 6 ft (2 m) apart. The generator is inserted into the 11-in (0.28 m)-long electrom air gap. Instrumentation is provided to measure and control gas and liquid flow rates, pressures, and temperatures; and to measure (dc) generator voltage and current, pressure along the generator channel, magnetic field strength, and void fraction distribution in the generator.

General Parameters

Working fluids	Sodium and nitrogen
Maximum flow rates	500 gpm (26 kg/s) sodium 324 scfm (0.18 kg/s) nitrogen
Maximum pressures	100 psia (0.69 MPa) sodium 145 psia (1 MPa)
Maximum temperature	1000°F (811 K)

AUXILIARY EQUIPMENT:

- Data acquisition system - up to 96 channels; temperature or voltage input; punched tape output.
- dc electromagnet, 0 to 0.9 T-working volume: 22 in. (0.56 m) by 12 in. (0.30 m) by 11 in. (0.28 m) gap.
- Void fraction measuring system, 3 channels.

BIBLIOGRAPHY:

1. "HT-LMMHD System Design Description," Argonne National Laboratory Document No. G0017-0157-SA-01, December 1978.

AMBIENT-TEMPERATURE LIQUID-METAL MHD FACILITY (AT-LMMHD)

ATION: ANL-E, Argonne, Illinois

ATUS: Operational

INCIPAL USES:

The Ambient-Temperature Liquid-Metal MHD Facility is used to evaluate two-phase MHD generator performance with NaK and nitrogen, and to study the flow behavior in the generator.

CILITY DESCRIPTION:

The facility provides known flows of liquid (NaK) and gas (nitrogen) for testing two-phase liquid-metal MHD generators at ambient temperature. It consists of a NaK loop and a nitrogen loop with a common mixer, generator test section, and separator/NaK storage tank. The NaK is recirculated, while the nitrogen is exhausted to the atmosphere from the separator tank through a secondary separator. The nitrogen exhaust line is equipped with a control valve and controller to sense and fix the generator exit pressure. The NaK loop includes three canned-rotor pumps in series with bypass valves so that only one or two can be operated if desired, a turbine flow meter, control valves, and pressure gauge. The nitrogen loop comprises a heater and control system to heat the gas to the liquid temperature, a flowmeter, control valves, and pressure gauges. A dc electromagnet provides the magnetic field.

EST CAPABILITY:

The test section fits between conventional flanges, and is inserted into the (adjustable) electromagent air gap. Instrumentation is provided to measure and control gas and liquid flow rates, pressures, and temperatures; and to measure (dc) generator voltage and current, pressure along the generator channel, magnetic field strength, and void fraction distribution along the generator.

General Parameters

Working fluids	NaK (sodium-potassium) and nitrogen
Maximum flow rate	330 gpm (18 kg/s) NaK 800 scfm (0.44 kg/s) nitrogen
Maximum pressures	275 psig (2.0 MPa)
Maximum temperature	130°F (328 K)

UXILIARY EQUIPMENT:

Instrumentation for recording experimental data.

dc electrogmagnet, 0 to greater than 1.2 T at 3.5 in. (0.09 m) air gap, working volume 16 in. (0.41 m) diameter by adjustable air-gap length.

Void-fraction measuring system.

Hot-film anemometry system.

BIBLIOGRAPHY:

1. M. Petrick, G. Fabris, R. Cole, R. Hantman, E. Pierson, and J. Cutting, "Experimental Two-Phase Liquid-Metal MHD Generator Program," ANL/ENG-76-04, Argonne National Laboratory, Nov. 1976.

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Argonne, Illinois 60439

CHEMICAL PROCESSING OF LIQUID LITHIUM FUSION REACTOR BLANKETS*

by

J. R. Weston, W. F. Calaway, R. M. Yonco
J. B. Hines†, and V. A. Maroni

March 1979

Prepared for the Proceedings of the 14th IECEC
August 5-10, 1979, Boston, MA.

*Work performed under the auspices of the U.S. Department of Energy.
†Student Aide from Eckerd College, St. Petersburg, FL.

CHEMICAL PROCESSING OF LIQUID LITHIUM FUSION REACTOR BLANKETS*

J. R. Weston, W. F. Calaway, R. M. Yonco, J. B. Hines†, and V. A. Maroni

Argonne National Laboratory
Chemical Engineering Division
9700 South Cass Avenue
Argonne, Illinois 60439

ABSTRACT

A 50-gallon-capacity lithium loop constructed mostly from 304L stainless steel has been operated for over 6000 hours at temperatures in the range from 360 to 480°C. This facility, the Lithium Processing Test Loop (LPTL), is being used to develop processing and monitoring technology for liquid lithium fusion reactor blankets. Results of tests of a molten-salt extraction method for removing impurities from liquid lithium have yielded remarkably good distribution coefficients for several of the more common nonmetallic elements found in lithium systems. In particular, the equilibrium volumetric distribution coefficients, D_v (concentration per unit volume of impurity in salt/concentration per unit volume of impurity in lithium), for hydrogen, deuterium, nitrogen and carbon are ∿3, ∿4, >10, ∿2, respectively. Other studies conducted with a smaller loop system, the Lithium Mini-Test Loop (LMTL), have shown that zirconium getter-trapping can be effectively used to remove selected impurities from flowing lithium.

THE CHEMICAL ENGINEERING DIVISION OF THE ARGONNE NATIONAL LABORATORY (ANL) is conducting a program of studies to establish the technology for processing and monitoring liquid lithium fusion reactor blankets. Emphasis during the initial phases of this work has been on control of the hydrogen isotopes and the primary ambient impurities, oxygen, nitrogen, and carbon. The long term objective is to develop processing methods capable of 1) holding O, N, C, H and, if necessary, D levels in large liquid lithium systems at ≤10 wppm and 2) recovering tritium with sufficient efficiency to hold the tritium level at or below 1 wppm. Methods currently under study include cold trapping, hot-getter trapping (using reactive metals), and molten-salt extraction. Most of the key experiments conducted to date have been performed using the 50-gallon-capacity Lithium Processing Test Loop (LPTL) and a much smaller loop the Lithium Mini-Test Loop (LMTL). The features of the LPTL and the LMTL that are important to this study are summarized in Table 1. Progress in this program up to and including calendar 1978 together with details of the experimental facilities has been discussed in a number of earlier publications (1-6)**. The purpose of this paper is to summarize the important features of work completed during the first half of calendar 1979. This work consisted mainly of 1) the first series of molten-salt extraction tests on the LPTL using LiF-LiCl-LiBr and 2) an analysis of the zirconium getter wires from the 10,000 hour LMTL test run (1,2).

*Work performed under the auspices of the U.S. Department of Energy.
**Numbers in parenthesis designate references at end of paper.
†Present address, Eckerd College, St. Petersburg, FL.

MOLTEN SALT EXTRACTION TESTS

The concept of using a molten salt extraction process to remove hydrogen isotopes and the ambient impurities from liquid lithium has been described in earlier publications (5,6). In brief, liquid lithium (specific gravity ∿0.5) is mixed with and separated from an all-lithium-halide molten salt (specific gravity ∿2.2). During mixing, the salt-like impurities (*e.g.*, LiH, LiT, Li_2C_2, Li_3N) are preferentially extracted from the lithium phase into the salt phase. After separation, the lithium is returned to the reactor and the salt is sent on to an electrolyzing tank, where the impurities are evolved (as H_2, T_2, N_2, CH_4, etc.) by low voltage electrolysis using a specially designed inert-gas-sparged electrode (6). The size of the associated processing equipment is determined by the tritium production rate, total lithium inventory, and allowable upper limits for tritium and impurities. In general, the volume of space, the power demand, and the cost of this processing equipment are not expected to be at all significant for presently conceived D-T burning fusion reactors (5,6).

Preparatory to the first series of molten-salt extraction tests on the LPTL, the 35-gallon-capacity LPTL salt tank, shown schematically in Fig. 1, was filled with ∿15 gallons of molten LiF-LiCl-LiBr (22-31-47 mol %, m.p. ∿445°C). Following several weeks of low-voltage pre-electrolysis of the salt to minimize residual impurities, ∿6 gallons of lithium was pumped from the LPTL reservoir tank into the inner annular region of the salt tank. Following an extended period of continuous stirring of the lithium/salt interface, filtered samples were taken from both the salt and metal phases in the salt tank. Selected samples from each phase were analyzed for H, D, N, and C to determine the equilibrium distribution coefficients for these impurities between the salt and metal. In addition, several lithium samples were analyzed for F, Cl, and Br to determine the amount of dissolved and/or entrained salt in the lithium which will subsequently be returned to the LPTL reservoir tank. Representative results of these analyses are summarized in Table 2. Information on the sampling and analytical methods is given in Table 3 and, in more detail, elsewhere (1, 7-9).

The data in Table 2 offer definitive evidence that the salt extraction method can provide the degree of processing capability needed to remove several of the dominant ambient impurities from a liquid lithium fusion reactor blanket. The volumetric distribution coefficients for nitrogen, and the hydrogen isotopes, >10 and >3, respectively, are particularly encouraging since no other removal methods for these elements have evolved to the point that a comparable demonstration of practicality and

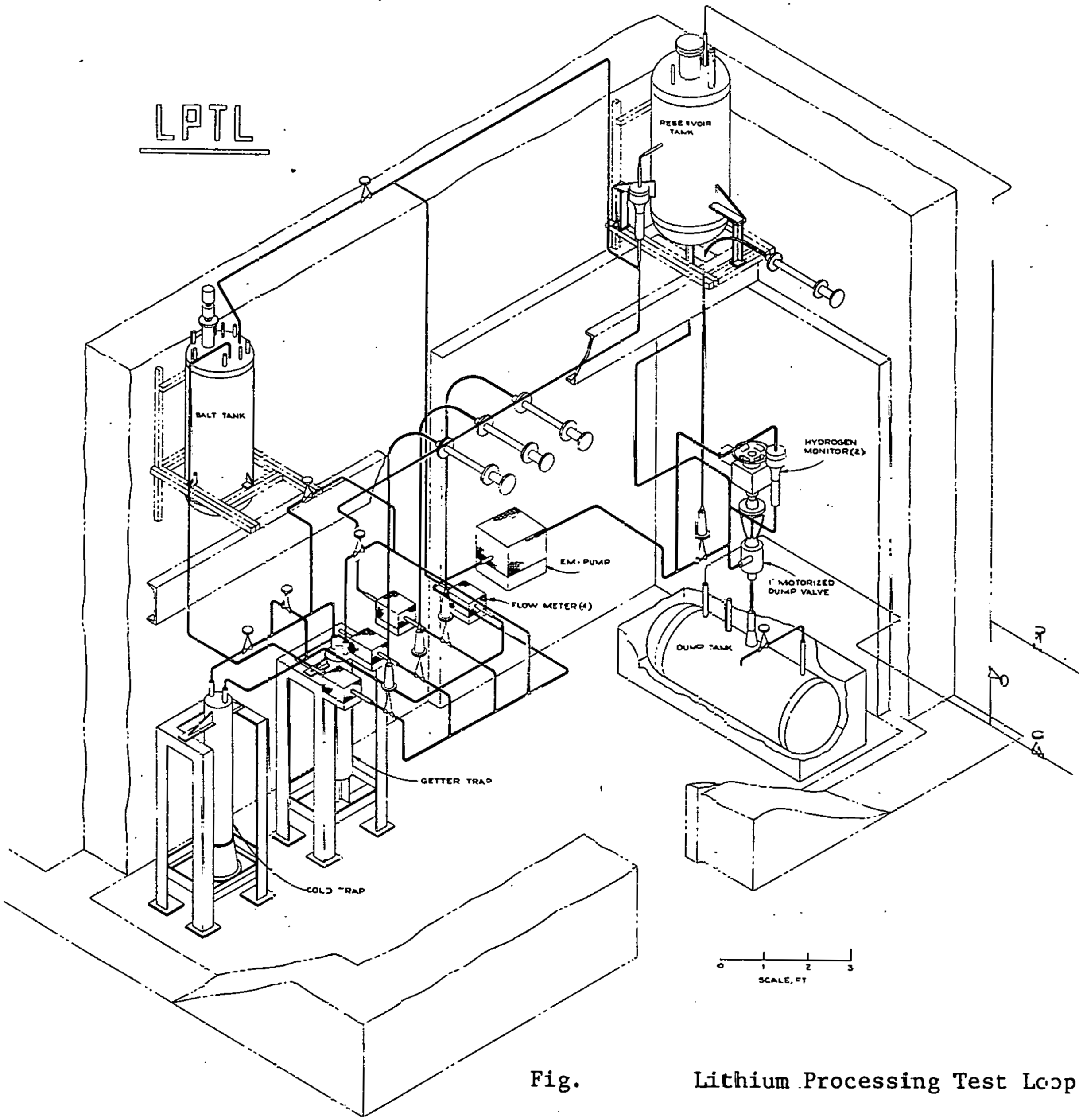

Fig. Lithium Processing Test Loop

Table 1 - Summary of Construction and Operating Characteristics for the Lithium Processing Test Loop and the Lithium Mini-Test Loop

	Lithium Processing Test Loop	Lithium Mini-Test Loop
Loop Capacity:	~50 gallons	~1 Liter
Construction Material:	Mostly 304L-SS (some 316-SS)	304- and 316-SS
Operating Temperature:		
Main Loop:	400 - 500°C	450 - 550°C
Getter Leg:	400 - 500°C	550 - 600°C
Cold Trap Leg:	180 - 300°C	—
Salt Extraction Leg:	500 - 530°C	—
Length of Operation:	6000 hours (continuing)	~10,000 hours
Experimental Capabilities:		
Direct Sampling:	Filter Pipette	Filter Pipette
Cold-Trapping:	Mesh-Packed Trap	—
Getter-Trapping:	Zirconium-Packed Trap	Zirconium-Packed Trap
Hydrogen Monitoring:	Permeation-Type	—
Salt Extraction:	LiF-LiCl-LiBr	—

Table 2 - Results of Salt-Extraction Tests Using the LPTL Salt Tank

(Test Temperature ≃530°C)

Phase Analyzed	Sampling Status	Impurity Element Concentration, wppm: Nitrogen	Hydrogen	Deuterium	Carbon	Chloride	Floride	Bromide
Lithium	Before Contacting	1200	800	8	-	550	15	50
Lithium[b]	After Contacting	IL-58	190	1.3	4.3	760	740	175
		IIL-36	-	-	2.2	490	980	270
Salt[b]	After Contacting	IS-261	130	1.1	1.9	-	-	-
		IIS-106	-	-	1.0	-	-	-
Volumetric Distribution Coefficient[a]		>10	3.1	3.9	2.0	-	-	-

[a]Volumetric Distribution Coefficient = $\frac{\text{(wppm impurity in salt)} \bullet \text{(density of salt)}}{\text{(wppm impurity in lithium)} \bullet \text{(density of lithium)}}$

Density of Lithium at 530°C = 0.483 g/cc; Density of Salt at 530°C = 2.2 g/cc

[b]Sample IL is correlated with sample IS, sample IIL is correlated with sample IIS.

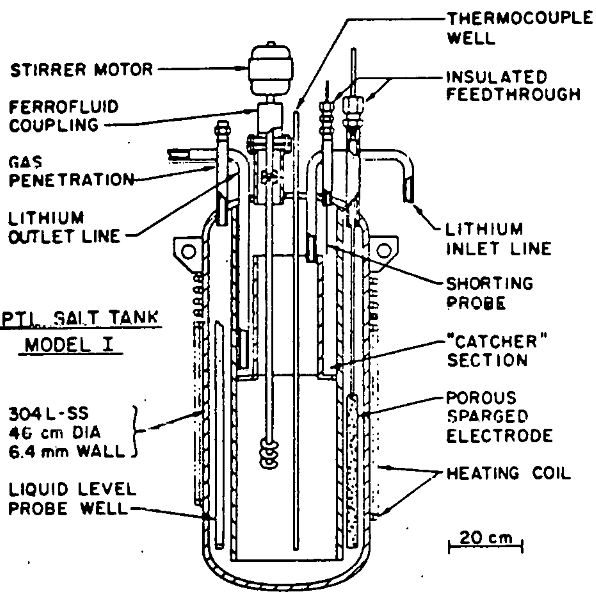

Fig. 1 - Schematic Cross-Section of the LPTL Salt Tank.

effectiveness can be made. Although the electrolytic evolution of nitride and hydride from lithium-saturated LiF-LiCl-LiBr has already been carried out on a laboratory scale (6, 10), further verification of this important step in the overall salt-extraction process is planned in subsequent tests on the LPTL salt tank.

While the total amount of halogen (F + Cl + Br) in the lithium following the salt/lithium separation step is near 2000 wppm, it is encouraging that only a small fraction of this is bromide. [Of the three halogens, F, Cl, and Br, only the bromide would pose measureable neutronic problems were it present in large amounts in lithium returning to a fusion reactor blanket.] Nonetheless, the salt processing branch of a reactor system will probably have to include a specially designed cold trap in the lithium line, downstream of the mixing and separating equipment, to reduce the total halogen content in a manner that

Table 3 - Summary of Sampling and Analytical Procedures Used on LPTL and LMTL

Impurity Element	Method	References
Hydrogen, Deuterium	Gas Evolution (Permeation)	1, 7
Nitrogen	Micro-Kjeldahl	1, 8
Carbon	Acetylene Evolution	1, 9
Fluoride, Chloride, Bromide	Ion Chromatography	-

would also allow for continuous or periodic return of the separated salt to the salt loop.

The principal of operation of the salt tank in Fig. 1 is that, following mixing and gravitational separation, the salt is pushed downward in the outer annular region (by controlled gas pressurization) until the salt/metal interface is within a few inches of the top of the "catcher", wherefrom it is forced back into the LPTL reservoir tank, *via* the lithium outlet line, by gas pressurization of the salt tank. A further function of the salt tank is that after the salt has become loaded with extracted impurities, it can be forced into the outer annular region which contains the same type of sparged processing electrode described in reference 6. The impurities evolved by the electrolysis (H_2, HD, D_2, N_2, etc.) are carried out of the salt tank by the sparge gas (high purity argon) and are gettered externally by a hot (~550°C) titanium bed.

Although the LPTL salt tank has proven to be especially well suited for determining equilibrium salt/metal extraction coefficients and related salt processing data, an interesting problem has emerged that has complicated efforts to carry out repetitive extraction/electrolysis tests. Over extended time periods, lithium from the inner annular region gradually migrates to the surface of the outer annular region and shorts the electrodes. The cause of this migration has not been clearly resolved, but thermal diffusion through the salt or some type of surface creep phenomena are suspected. In order to overcome this problem, a small auxiliary tank has been set up above the salt tank to provide a means of syphoning the migrating lithium from the outer annular region back into the inner annular region. This experience has confirmed earlier beliefs that, in the prototype salt processing scheme, mixing, separating, and salt electrolysis should be carried out in at least two or possibly three separate and totally independent steps.

LONG-TERM GETTERING STUDIES USING ZIRCONIUM

Hot-getter trapping, using reactive metals, has long been considered a viable approach to the problem of removing the ambient impurities from liquid lithium (11,12). Accordingly, a Lithium Mini-Test Loop (LMTL) experiment was initiated at ANL in 1976 to evaluate thermally regenerative hot trapping by zirconium. The basic features of the LMTL experiment are summarized in Table 1 and are discussed in greater detail in references 1 and 2.

In all, the LMTL operated for ~10,000 hours before shutdown. During the first ~1000 hours, the nitrogen level dropped from the startup value of ~700 wppm to ~200 wppm, but thereafter rose linearly with time to ~600 wppm at 10,000 hours (2). The carbon level, determined near the end of the test, was <50 wppm. As part of the post-test examination of the LMTL, the zirconium wires (~1.5 mm in diameter) which had been used in the thermally-regenerative getter leg, were subjected to analysis by ion microprobe (IMMA) methods (13). The results of those analyses are given in Table 4 and Fig. 2.

Microscopic examination of selected wires mounted in cross section, revealed that the first 15 to 30 μm region (inward from the surface) was heavily pitted and cracked and was largely inhomogeneous. The data in Table 4 are considered to be representative of IMMA spot scans made across this region. They show that the surface region contains measureable quantities of the constituent elements of the loop construction material (Fe, Cr, and Ni) and significant quantities of the major alkali and alkaline earth elements present as impurities in the original lithium (*i.e.*, Ca, Na and K). The latter elements are presumed to be tied up as ternary or higher order zirconium compounds

Table 4 - Metallic Element Distributions[a] in Zirconium Getter Wire from 10,000 Hour LMTL Test

Element	Relative Counts (Averaged) 0 to 30 μm	Relative Counts (Averaged) >30 μm (Bulk)
Zirconium	~4 x 10^4	~6 x 10^4
Iron	~4000	<200
Chromium	~ 300	<100
Nickel	~1000	<100
Calcium	~1 x 10^4	<100
Lithium	~10^5	<100
Potassium	~2 x 10^4	<100
Sodium	~5000	<100

[a]Determined by Ion Microprobe Mass Analysis

such as zirconates. This supposition remains to be tested by more specific methods (*e.g.*, Auger or ESCA). If true, however, it would appear that, whereas reactive metal gettering with materials like zirconium might provide some capability for removing the alkali and alkaline-earth elements from lithium, the compounds of these elements formed around the near-surface regions could reduce the effectiveness of the getter in extracting the non-metallic elements. Although not indicated in Table 4, it was also found that the relative concentrations of H, O, N, and C in the near-surface regions (0 to ~30 μm) are much higher than in the bulk (>30 μm).

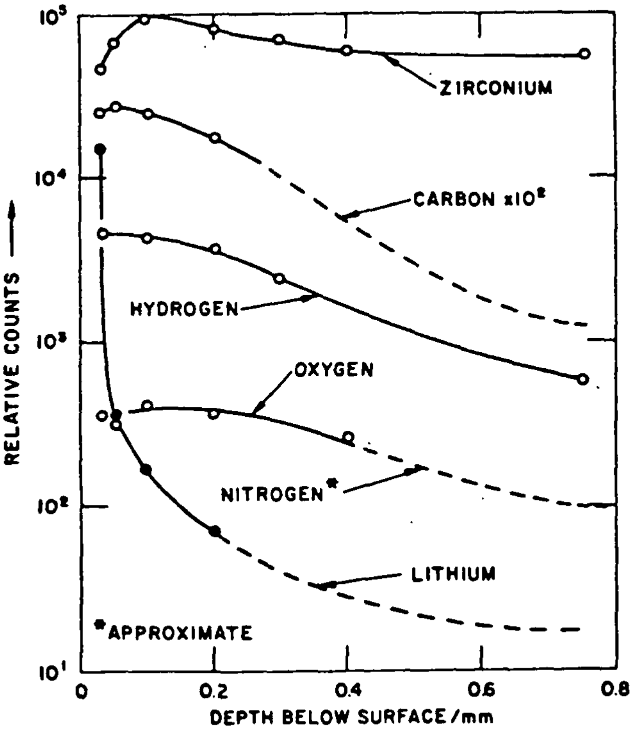

Fig. 2 - Ion Microprobe Analysis of Zirconium Wire From the 10,000 Hour LMTL Test.

The results in Fig. 2 give evidence that H, C, O, and N did indeed penetrate the getter wires (most likely by bulk diffusion) to depths of ≥400 μm. The estimated concentration of each of these elements in the wire, based on crude integration of the extrapolated profiles, is ~0.1 weight percent, which in any future usage should be correlated with the fact that the wires were operated in lithium for 10,000 hours at 600°C and that the specific surface area of the wires was ~4 cm^2/g. It should also be stated here that the data for nitrogen were somewhat obscured by other mass signals and that the profile for nitrogen was estimated to be about the same as the oxygen profile as is indicated in Fig. 2. In the case of O, N, and C, the dashed portions of the curves in Fig. 2 represent upper limits to the extended profile and, in fact, in each case the center reading (at 0.75 mm) was actually below the limit of reliable detection (~80 counts). The apparently higher concentration of hydrogen in the 0 to 400 μm region, compared to the wire-centerline value (at 0.75 mm), may be due to trapping of hydrogen by other impurities or lattice defects (14,15). Ordinarily, the hydrogen profile would be expected to be relatively flat, because of the high diffusivity of hydrogen in zirconium at 550-600°C.

CONCLUSIONS REGARDING LITHIUM PROCESSING CAPABILITIES

Based on the work reported herein and in reference 1, 2 and 6, it is possible to define in reasonably specific terms a processing methodology for liquid lithium that will meet the needs of inertial and magnetic fusion power reactors. A summary of projected processing capabilities for the three purification methods (cold trapping, getter trapping, and salt extraction) studied in this paper and in reference 1 is given in Table 5. These results indicate that a combination of salt extraction and cold trapping should provide adequate tritium recovery and impurity control. It is expected that the hydrogen isotopes, nitrogen, and to some extent carbon will be removed *via* the salt extraction equipment and that cold trapping will control carbon, oxygen and halogen levels.

Because the salt-extraction step must be carried out at 500 to 550°C, it will probably have to be performed on the lithium at the peak temperature of the thermal conversion cycle. For the case of lithium cooled reactors, it will be incumbent upon future process verification tests to determine whether the removal from lithium of entrained and dissolved salt must be carried out immediately after the salt processing operation or can be delayed until after the energy extraction step. Since only a small fraction of the total lithium flow coming from the reactor would be directed to the salt processing leg (1 to 2% perhaps), the loss of thermal energy accompanying cold trapping of the process stream (before energy extraction) would not be severe, and, in fact, some of that energy might be recoverable through clever cold-trap economizer design. On the other hand, it seems that no serious problems would arise from blending the process stream in with the primary entrant flow to the intermediate heat exchanger (IHX) and then cold trapping some fraction of the low temperature lithium returning to the reactor from the IHX. Regardless of which approach is used, the salt level in the bulk lithium will eventually reach the level set by the effective cold-trap temperature. This level appears to be <300 wppm total halogen, Cl + Br + F, based on preliminary results of some cold trap experiments that are still in progress.

In an actual reactor blanket system, it should be possible to reduce the steady-state concentration of the hydrogen isotopes (total) and the ambient impurities (excluding oxygen) to the ≤10 wppm level. By permitting the hydrogen and/or deuterium steady-state concentrations to exceed the tritium concentration by a factor of 10 or more, the tritium level can be held to <1 wppm. This does not appear to put an unreasonable burden on subsequent fuel enrichment operations. More work is needed to better define a cold-trapping or gettering procedure that will effectively reduce oxygen to the <10 wppm level. Because the decomposition potential of Li_2O is higher than that of the extraction salt (LiF-LiCl-LiBr), it will not likely be possible to evolve oxygen during salt electrolysis.

On the whole, the prospects for maintaining a low-tritium-inventory, high-purity circulating lithium system in a fusion power reactor appear to be quite good at this time. Steps are presently being taken in the ANL blanket processing program to provide a definitive verification of dynamic real-time tritium recovery and simultaneous impurity control on a scale that should extrapolate to anticipated fusion reactor operating conditions.

Table 5 - Summary Comparison of Processing Options

	IMPURITY CONTROL METHODS[a]		
IMPURITY	COLD-TRAPPING	GETTER-TRAPPING	SALT EXTRACTION
TRITIUM	>1000 wppm	(c)	<1 wppm
NITROGEN	>1000 wppm	(c)	<10 wppm
CARBON	<5 wppm	(c)	<10 wppm
OXYGEN	≥100 wppm[b]	(c)	(d)

[a]Numbers indicate projected upper or lower limit of control.

[b]Without post-trap filtration.

[c]Depends on getter material, temperature, and trap design-- <100 wppm is possible.

[d]Not applicable.

ACKNOWLEDGEMENTS

The authors are indebted to C. E. Johnson and D. V. Steidl for performing the ion microprobe (IMMA) analyses, to M. I. Homa for assistance in the carbon analyses, to R. Crooks for performing the nitrogen analyses, and to E. Veleckis for assistance in the hydrogen/deuterium analyses. The support and encouragement of L. Burris, Jr, F. A. Cafasso, F. E. Coffman, and J. E. Baublitz are gratefully acknowledged. This work was sponsored by the U.S. Department of Energy/Office of Fusion Energy.

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U.S. Department of Energy
ARGONNE NATIONAL LABORATORY
9700 South Cass Avenue, Argonne, Illinois 60439
Telephone 312/972-5317

November 20, 1981
Ref: TPE 11-26

Mr. Glen Longhurst
EG&G Idaho, Inc.
Idaho National Engineering
Laboratory
P.O. Box 1625
Idaho Falls, Idaho 83401

Dear Glen:

The attached description of ANL Bldg. 206 Sodium Loop Facilities should be added to the package on this topic that I recently sent to you.

Yours truly,

H. Herman
Deputy Manager
First Wall/Blanket/Shield
Program

HH:mlw
Enclosure

cc: B. W. Spencer - RAS

Table 1. SODIUM LOOP FACILITIES AT ANL, BLDG. 206

	[a]CAMEL-II	[b]OPERA	[c]SSL
Facility Configuration	Loop or Blowdown	Blowdown	Loop
Fluid	Sodium	Sodium	Sodium
Heat Source	Trace Heating	Trace Heating & Fuel Pin Simulators	Trace Heating
Heat Sink	Heat Capacity	Heat Capacity	375 kw Heat Exc.
Design Temperature, °F	1000	1200	1200
Design Pressure, psig	300	200 (Blowdown Vessel) 10 (Receiver Vessel)	470
Flow Control	EM Pump (Loop) or Pressure Differential	Pressure Differential	Annular - Linear Induction Pump
Sodium Flowrate, gpm	220 (Loop) 400 (Blowdown)	250	120
Pump Head, psid	150	–	120
Sodium Inventory in Loop/ Total, Gal.	60/80	65	20/50
Test Section(s)	9.6 ft. Vertical, 3 ft. Horizontal	12 ft. Vertical	6 ft. Vertical

[a]Components and Materials Evaluation Loop

[b]Out-of-Pile Expulsion and Reentry Apparatus

[c]Small Sodium Test Loop

Argonne brochures "Components Technology" and "Argonne Facilities" are not included since they contain numerous photographs which could not be effectively reproduced in this report. These brochures may be obtained from EG&G Idaho Division of Fusion Technology.

APPENDIX-M

APPENDIX M

LOS ALAMOS NATIONAL LABORATORY
RESPONSE

University of California

LOS ALAMOS SCIENTIFIC LABORATORY

Post Office Box 1663 Los Alamos, New Mexico 87545

In reply refer to: CTR-12
Mail stop: 641

October 1,1981

P. Y. Hsu
TPE-II Program Manager
Fusion Technology Program
EG&G, Idaho, Inc.
P. O. Box 1625
Idaho Falls, Idaho 83415

Dear Mr. Hsu:

I am replying to your letter of September 11, 1981, requesting responses to questions concerning Los Alamos capabilities in performing portions of TPE-II of the overall ANL FW/B/S program.

Los Alamos responded to the original EOI for TPE-I and TPE-III. The material submitted as part of the Los Alamos EOI response to TPE-I can be applied directly to the bulk of the TPE-II tasks. It is within this context that I enclose a copy of the Los Alamos TPE-I response.

In surveying the subquestions under items #1 and #2 of your questionnaire, we feel that all non-nuclear requirements of TPE-II can be met or exceeded by the facilities, personnel and/or general capabilities outlined in the enclosed Los Alamos response to EOI/TPE-I. The major sources of steady-state nuclear (neutron) heating at the Laboratory would be a small pool-type fission reactor (Omega West) and the beam-dump at the LAMPF. Relevant responses to your item #3 of the questionnaire with respect to the Omega West Reactor and the LAMPF beam-dump can be generated at a later date, should you so desire.

Thank you for your interest, and please do not hesitate to call for added information. (FTS-843-5863).

Sincerely yours,

Robert A. Krakowski

Robert A. Krakowski

RAK/odm

encls. a/s
xc: CRMO (2)
File

RESPONSE TO REQUEST FOR EXPRESSION OF INTEREST

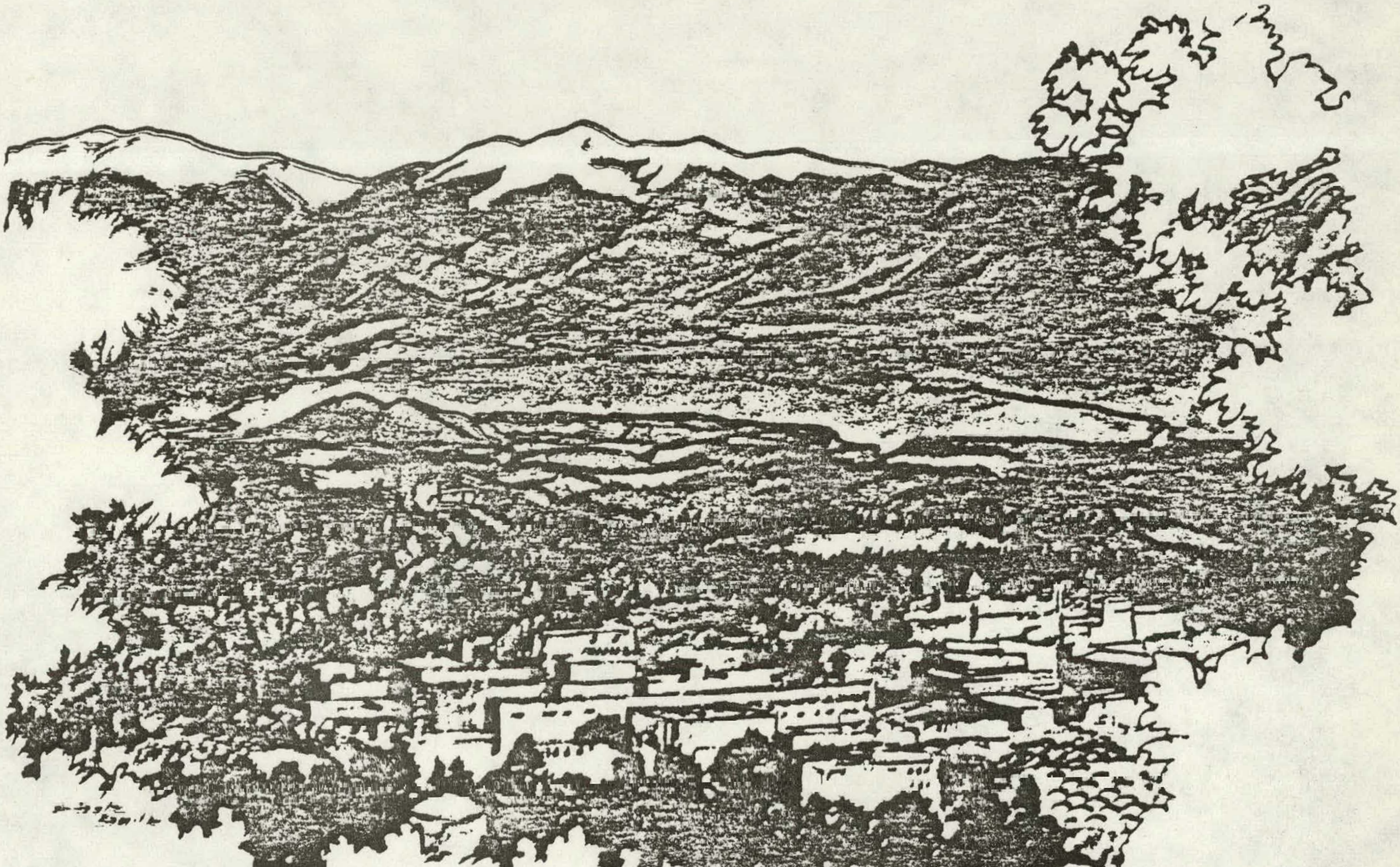
(EOI) TO CONDUCT TEST PROGRAM ELEMENT I

(TPE-I) OF FIRST-WALL/BLANKET/SHIELD (FW/B/S)ENGINEERING

EVALUATIONS/TESTS

Submitted by

LOS ALAMOS NATIONAL LABORATORY



LOS ALAMOS SCIENTIFIC LABORATORY

RESPONSE TO REQUEST FOR EXPRESSION OF INTEREST (EOI)
TO CONDUCT TEST PROGRAM ELEMENT I (TPE-I) OF FIRST-WALL/BLANKET/SHIELD
(FW/B/S) ENGINEERING EVALUATIONS/TESTS

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  B. TECHNICAL BASIS
  C. GENERAL ORGANIZATION OF TPE-I
  D. LOS ALAMOS ORGANIZATION FOR IMPLEMENTATION OF TPE-I.
II. RESPONSE TO REQUEST FOR EOI ON TPE-I
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      b. Transient Effects
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  E. PHASE I SCHEDULE
  F. PRINCIPAL AND KEY INVESTIGATORS

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ATTACHMENT #2. APPROACH TO TPE-I(PHASE I)
ATTACHMENT #3. TPE-I SCHEDULE
ATTACHMENT #4. PARTIAL LIST OF COMPUTER PROGRAMS IN USE AT LOS ALAMOS WITH APPLICABILITY TO TPE-I
ATTACHMENT #5. GENERAL DESCRIPTION OF EXISTING TEST FACILITY TO BE USED FOR TPE-I
ATTACHMENT #6. RESUMES OF KEY INVESTIGATORS

I. INTRODUCTION/BACKGROUND

This section summarizes the essential elements of TPE-I as given by the "Program Plan for DOE/Office of Fusion Energy First-Wall/Blanket/Shield Engineering Test Program" (November, 1980).

A. DEFINITION OF TPE-I

The task definition for TPE-I is as follows: non-nuclear thermalhydraulic and thermomechanical testing of first-wall and, where integrally coupled, first-wall/blanket facsimiles with emphasis on surface heat load and heat-load transient (i.e., plasma disruption) effects.

B. TECHNICAL BASIS

The experimental program for TPE-I addresses a variety of first-wall and first-wall/blanket configurations under normal and transient heat-loadings. Specifically, these tests will:

- examine the first-wall component reliability and integrity under conditions of sensible heat recovery with fusion-relevant, combined hydraulic and mechanical responses.
- model experimentally the hydraulic and mechanical response of fully-structured first walls (i.e., including bends, weldments, fasteners, supports, coatings) under transient condition.
- perform engineering tests to failure *via* single- or multiple-pulse operation of realistically engineered first-wall structures to develop a design basis for reliable, cost-effective first-wall configurations.
- examine the hydraulic and mechanical response of specialized first-wall components (pumped-limiters, coatings/liners, protective armor) that are expected to be subjected to both steady-state and transient surface heat loads that far exceed the design limits *per se* anticipated for the first wall.

C. GENERAL ORGANIZATION OF TPE-I

Phase 0: Detailed planning of TPE-I that factors in the developing definition of needs for ETF/FED.

Phase I: Development of heat ejector(s) that gives a realistic simulation of operational effect on a representative first-wall and/or first-wall/blanket segment under fusion-specific hydraulic and mechanical simulations. The general objectives of TPE-I are:

- develop and operate a first-wall test facility that can deliver heat at a rate of 1-2 MW/m^2 to first-wall segments of area $\geq$ 0.1 m^2, 10-20 MW/m^2 to small components, and as high as 100 MW/m^2 under transient conditions.
- examine a range of first-wall configurations, including panel coils, tube-banks, and slotted monoliths of various thicknesses, shape, materials, coolants and coolant parameters, structural supports, coatings, and temperatures.
- examine under high heat flux a range of first-wall components, such as limiters and protective armor.
- evaluate thermal and mechanical operating limits, identify failure mode and optimal configurations for a range of first-wall and first-wall component configurations for both steady-state and transient conditions.
- develop a first-wall engineering data base through comparison of experimental results with numerical models that can be used by or extended to ETF/FED conditions insofar as stress, strain, fatigue, deflections, deformations, vibrations, failure modes, effects of failure and general thermohydraulic responses are concerned.

D. LOS ALAMOS ORGANIZATION FOR IMPLEMENTATION OF TPE-I.

The general organization of Los Alamos staff proposed to undertake TPE-I and to achieve meaningful experimental results in the shortest possible time on existing facilities is shown in Attachment #1. The centrum of the experimental effort will be located in the Advanced Heat-Transfer Group Q-13. Thermohydraulic and thermomechanical analyses for both design and interpretation of all experiments also will be performed in Group Q-13, with significant support and guidance being provided by Groups CTR-12 and WX-4 in order to assure the optimal generation of fusion-relevant data and data correlations.

II. RESPONSE TO REQUEST FOR EOI ON TPE-I

A specific response to the request for EOI on TPE-I is contained herein. This response follows explicitly the outline suggested by the ANL request-for-EOI memorandum of November 24, 1980 (Sec. II.A: Approach; Sec. II.B: Integration; Sec. II.C: Characteristics; Sec. II.D: Support; Sec. II.E: Schedule; Sec. II.F: Principals). The intent of this response is to demonstrate the unique combination of existing thermal test facilities, advanced-heat-transfer expertise and fusion-relevent experience that presently resides and is active at Los Alamos; the timescale requested for this EOI does not permit the planning of a specific, extensive and quantitative experimental program to be included, the inference being that such activity would be included in the Phase 0 part of TPE-I.

A. TPE-I(Phase I) Approach

1. General Experimental Approach/Objectives. The general approach proposed for TPE-I is illustrated schematically on Attachment #2. This well-tried and successful approach is based on the utilization of existing facilities, including a 2.5-MW dc generator, well-equipped test bays, large vacuum chambers and modern data acquisition systems. The proposed tests will be performed in conjunction with state-of-the-art thermomechanical and thermohydraulic analyses and codes to produce fusion-relevant results under the close guidance and/or direct participation of in-house experts in the areas of advanced heat transfer, mechanical design/analysis and fusion technology. Meaningful results should be forthcoming within six to nine months of the project initiation. Because of the existence and availability of both impressive facilities and highly-qualified staff, a wide range of first-wall configurations and first-wall components can be tested, analyzed and correlated with a rapid and inexpensive retrofit capability already in place and functioning.

Specifically, all first-wall and first-wall component tests will be performed in vacuo or in low-pressure hydrogen (few mtorr) in order to simulate closely the fusion-device environment. Focused, controllable and programmable radiant heaters will be used to deliver as much as 2 MW of dc power to first-wall segments that are of fusion-relevant size ($0.1-1.0\ m^2$). Auxiliary power sources and heating methods (e.g., electron beams, resistance heating) are available to superpose on any first-wall test segment localized heat fluxes in excess of $10-20\ MW/m^2$. Both additional space and power is available to provide neutral-beam or ion-beam heating of large surfaces, given that the acquisition of such heaters is deemed desirable during the Phase II activity. The use of plasma arc-jet heating, either as the main or auxiliary surface-heating scheme could also be used, although this scheme was rejected as not representative of the fusion environment (i.e., because of high pressure). The versatility of the main dc power source operating in conjunction with numerous, large test bays permits as many as four major tests to be performed simultaneously, should the funding for TPE-I ever match this unique experimental/testing capability.

The overall program objectives, as outlined in the "DOE/Office of Fusion Energy First-Wall/Blanket/Shield Engineering Test Program Plan" and as summarized in Secs. I.B. and I.C. of this EOI are straightforward and appropriate. Implementation of the objectives on the basis of a qualitative interpretation must await the results of the Phase-0 activities. Generally, the development of an appropriately configured and cooled first-wall segment of 0.1-1.0 m^2 area and the transfer to that test segment a steady-state, uniform heat flux of 1-2 MW/m^2 represents a state-of-the-art task. Although long-term steady-state tests under these conditions and in this environment will be performed to extend an already substantial data basis, the focus of the TPE-I activities will be placed on failure mode analysis and component evaluation under more severe pulsed, higher-heat-flux conditions and on heat-transfer surfaces that are not conventional (i.e., coated surfaces, brittle materials, test elements that have the potential for local failure points, such as weldments and bends). In the processes of developing and extending the engineering design data base for these less than "ideal" conditions, all experiments will be designed and verified by state-of-the-art computer codes and models. As noted previously, these tests will apply an auxiliary, localized power density to develop "hot-spot" conditions that will force the kinds of structural/coolant interactions expected to occur in a fusion engineering system, even under conditions of "normal" operation.

2. Program Structure. The organizational structure that will be implemented at Los Alamos to assure the successful execution of TPE-I has been shown schematically in Attachement #1. Although the TPE-I experiments *per se* will be designed, operated and analyzed primarily by the Advanced Heat-Transfer Group, Q-13, the overall program will be advised and guided from the fusion technology/systems-studies group, CTR-12. A Los Alamos advisory committee will be formed to assure the fusion relevance of TPE-I, this committee being comprised of experts in the area of fusion technology/systems-studies, heat transfer, thermomechanical design/analysis and materials. Provisions for optimal facility usage and program growth will be assured through the TPE-I advisory committee, this committee representing the primary contact with ANL through an appropriately appointed program manager. As shown on Attachment #1,

the program manager would also be responsible for other FW/B/S TPE's, should they come to Los Alamos.

3. Program Schedule. Attachment #3 gives a tentative program schedule for TPE-I. As noted previously, the unique, high-power facility coupled with an experienced and diverse support staff promises a rapid start and a flexible program. As presently structured, the funding level for TPE-I is personnel limited rather than facility limited, thereby assuring a strong potential for rapid and cost-effective growth in test program output.

B. RELEVANT EXPERTISE AND INTEGRATION WITH OTHER FUSION EFFORTS

1. Fusion-Specific Physics. The integration of past and developing fusion-relevant physics (i.e., character and level of heat fluxes and volumetric sources, first-wall and first-wall-component configurations, transient responses) will be assured through the TPE-I advisory committee by Group CTR-12. Group CTR-12 has a strong background in a wide range of fusion technology and physics issues, works with the support of a wide range of plasma physics experts within CTR-division, and has worked with other groups on programmatic tasks.

2. Relevant Engineering and Testing Experience. Los Alamos provides a coordinated and experienced engineering design team (Group WX-4) in the Design Engineering Division, WX, that specializes in both systems and component engineering. For all major scientific projects conducted in the Laboratory the responsibilities of WX-Division include the provision of direct feedback to systems designers, including designers of conceptual magnetic and inertial fusion reactors. Engineering projects in which the Design Engineering Division has been associated include many first-of-a-kind systems that have advanced the state of technology in laser fusion, magnetic fusion and high-energy physics. Group WX-4, specifically, has contributed significantly to the Weapons Neutron Research (WNR) facility, the Toroidal Reversed-field Pinch (ZT-40) experiment, the Helios and Antares CO_2-laser test facilities and the Los Alamos Meson Physics Facility (LAMPF). Engineering tests with relevance to TPE-I that have been performed include heat-transfer and thermal-stress analyses of magnets, first walls, targets and energetic-particle-beam dumps. Engineering design projects have included remote handling systems, large vacuum systems, precision target insertion/recovery mechanisms, particle accelerating cavities, rf power generators, magnet stands, alignment fixtures and blast containment vessels.

In projects that are directly applicable to TPE-I, heat-transfer, fluid-flow and structural analyses have been conducted in the course of conceptual design of fusion reactor first-walls and blanket systems for the following magnetic fusion reactor concepts: Reference Theta-Pinch Reactor, Reversed-Field Pinch Reactor, Elmo Bumpy Torus Reactor, Linear Theta Pinch Reactor and the Tandem Mirror Reactor. Additionally, conceptual designs have been completed for laser-driven inertially-confined fusion reactors, utilizing both wetted and magnetically-protected first walls, and for the Imploding Fast-Liner Reactor. References 1-6 give an example of past fusion-specific work performed by Group WX-4. The combination of analysis and design skills developed by Group WX-4 in these prior programs provide a unique basis of experience that is directly relatable to the tasks of developing, planning and executing TPE-I.

Directly relatable testing experience has also been developed in the successful and timely completion of a "proof-of-principle" experiment for evaluation of programmed electron-beam heating as a means to simulate first-wall thermal and mechanical environments.[7] This experience provides a basis for using electron-beam heating as a supplement to the aforementioned radiant heat flux method proposed as the mainstay for the TPE-I experimental activity. Additionally, experience with large, high-current accelerators, such as LAMPF and the FMIT ion source, provides a strong background for evaluation and development of future, comprehensive first-wall simulations, including effects of ion-bombardment and plasma/wall interactions. Examples include a test facility presently being used to determine the effects of cyclic stress on metallic samples undergoing simultaneous 800-MeV proton irradiation (LAMPF).[8]

Test experience in the simulation of thermohydraulic effects in loss-of-coolant and loss-of-flow accident conditions for the gas-cooled fast reactor (GCFR) provides additional background at Los Alamos with relevance to TPE-I. This semi-scale test program was conducted by Group Q-13 over a four-year period and utilized resistance-heating simulations of nuclear heating in appropriate fission-fuel arrays. The experience developed in high-power, electrical system control, hydraulic instrumentation and data processing, as well as with the test facilities themselves, are all directly applicable to TPE-I. These thermohydraulic/thermomechanical test facilities are described in Sec. C.3.

Much of this testing experience, which has been developed under conditions that generally exceed in difficulty those anticipated for TPE-I, originated with the nuclear rocket program in the 1960's and has continued since. Hence, the Los Alamos team would bring to the FW/B/S Test/Evaluation Program over 20 years of testing/evaluation experience that strongly emphasizes the engineering data-base development for advanced-technology systems.

3. Relevant Analytic Modeling Experience. An extensive background in analytic modeling of thermomechanical and thermohydraulic responses of advanced engineering systems has been developed at Los Alamos through prior fission-reactor safety studies and high-energy physics research projects. More recent thermomechanical model development includes a transient thermoelastic analysis of the graphite core-support structure of the High-Temperature Gas-Cooled Reactor (HTGR) that has been subjected to a severe cool-down thermal transient following a loss-of-forced-circulation accident.[9] Another recent study in this area analyzed the transient response of a sapphire window used to view reactor core disruptions during the accident sequence; the results of this analysis gave recommendations that when applied precluded window failure through thermal shock. Low-stress fin designs for high-performance ceramic heat exchanges have also resulted from analysis of transient thermoelastic stresses, this activity also serving as another example of an activity that is directly relatable to the analysis requirements anticipated for TPE-I.

Relevant analytical experience also includes the calculation of the transient response of fission-reactor pressure vessels and containment to pressure buildup.[10] These analyses use nonlinear methods for the estimate of ultimate structural capabilities. The correlation of model predictions with substantiating experiments has also been performed for the determination of buckling loads of fission reactor containment structures. Analytical techniques for the prediction of creep have been incorporated into existing computer codes and have been applied to a wide range of light-water fission reactor safety problems. Recent analytical techniques have been developed to describe fluid/structural/thermal interactions[11] related to loss-of-flow accidents in the GCFR and relevant test programs being conducted at Los Alamos. A number of the finite-element codes used to describe these problems could be or have been modified to include electromagnetic forces and other related magnetic field interactions.

Attachment #4 gives a partial list of computer programs presently used at Los Alamos with a high potential for application to the FW/B/S test/evolution program: The SPAR, TSASS, ADINA, and ADINA-T computer programs are of most interest to TPE-I. SPAR[12] is an interactive/batch computer code used primarily for performing stress, buckling and vibrational analyses of linear finite-element geometries. Sparse matrix solution methods provide a large-size capacity (> 2000 degrees of freedom), and a sophisticated data handling routine and arithmetic utility system makes the SPAR code especially useful for the manipulation of large data blocks. It is expected that this code system would be used extensively in the TPE-I design and data analysis activities.

The TSAAS code[13] is a two-dimensional finite-element computer program that is capable of performing both steady-state and transient thermoelastic analyses of multi-material systems. TSAAS is used for determining temperatures, displacements and stresses in complex solid-body configurations with orthotropic, temperature-dependent material properties. Non-linear mechanical behavior, as is typified by plastic, locking, tensionless, or creeping materials can be approximated. Because of its conciseness and coding simplicity, TSAAS is especially appropriate for developing new materials models and for this reason would be a mainstay analytic tool for TPE-I. The conciseness of TSAAS also lends this code to ready use as a real-time predictor in connection with the experimental test program. TSAAS has been applied to examine the transient response of complex, pulse magnets in connection with the design of fusion devices.[14]

ADINAT is a finite-element computer code[15] designed for linear and nonlinear steady-state and transient analysis of heat-transfer and other field problems. Treating conduction, convection and radiation in three dimensions, ADINAT stores temperature fields in a format that is easily read by the sister stress code, ADINA. Several linear and nonlinear conduction models are available. Temperature-dependent convection coefficients and emissivities can be used. Anisotropies in conduction properties can be treated as orthotropic constant conductivities. ADINA is a finite-element computer code[16] for linear and nonlinear, static and dynamic analysis of structures. It can be used to perform displacement and stress analysis of solids, structured systems, and fluid-structure systems. The finite-element library includes a wide variety of elements ranging from a three-dimensional truss to a three-dimensional solid element. Several material models are available. The more important models for use in the first-wall studies include an isotropic thermoelastic model and a curve description model with tension cutoff (cracking). ADINA is designed to accept data from the companion heat-transfer code ADINAT.

In order to take full advantage of these large finite-element computer codes a number of pre- and post-processors are available. The pre-processors typically are used for automatically generating meshes and loads from a minimum of input information. To manipulate and analyze the large quantities of data generated by these codes, particularly for transient analyses, the post-processors present data in a number of formats ranging from simple line plots to colored movies. Complex use of these pre- and post-processors will be of particular value in the thermomechanical analyses anticipated for the FW/B/S test/evaluation program.

C. TEST PROGRAM DESCRIPTION AND CHARACTERISTICS TO BE EVALUATED BY TPE-I (PHASE I)

1. Definition of Test Environment. The test environment for TPE-I will as a minimum meet the general guidelines given in the DOE/Office of Fusion Energy First-Wall/Blanket/Shield Engineering Test Program Plan and summarized in Secs. I.B and I.C. The operating environment will be vacuum or low-pressure (few mtorr) hydrogen, and the primary heater will be controllable radiant elements operating at a total steady-state power of a few megawatts. Typical operational test parameters will be:

- steady-state surface heat flux : 1-2 MW/m^2
- pulsed or concentrated heat fluxes: 10-20 MW/m^2
- highly-concentrated and destructive heat fluxes : > 100 MW/m^2
- typical test sample size : 0.1-1.0 m^2
- typical steady-state operating temperatures: : 500-800 K
- coolants : water or helium gas, both pressurized.

2. Test Element Selection. A typical test element will be 0.1-1.0 m^2 in area and will be formed from panel coils, tube arrays or slotted monoliths. The fabrication, shapes, appendages, weldments, fasteners, and materials will be selected and arranged to be prototypical of a range of fusion reactor first-wall concepts. Both coated and uncoated stainless steel will be used primarily, although a wide range of materials will be used under conditions of high steady-state and pulsed heat fluxes to simulate the high-performance responses of special first-wall components (e.g., limiters, protective armor). In later phases of TPE-I, first-wall ducts and penetrations will be fabricated

into the larger test items in order to examine special cooling problems that may be encountered in and around nonaxisymmetric fixtures related to the first wall, including electrical insulators. Local, nonsymmetric heating will be applied to these first-wall regions by electron-beam heaters (surface flux) or local joule heaters (bulk heating).

3. Test Facility Development

a. Summary of Requirements. The facility requirements for accomplishment of the test objectives of TPE-I(Phase I) as a minimum consists of a vacuum test enclosure capable of containing a > 0.1 m^2 area first-wall test with associated instrumentation, heating and cooling systems, approximately 2 MW of power supply for the heating system, an equal cooling capacity for water or helium coolant loops, and support facilities for test set-up and operation. In addition, provision must be made to modify the basic test facility in order to test smaller first-wall sizes at higher power densities. Fulfillment of the Phase I program within the proposed schedule will require use of an existing test facility, since the construction of a new facility would not be possible within temporal and monetary constraints imposed. Control systems for heating and cooling of the first-wall test should allow the imposition of programmed thermal flux shapes on the test array. Automatic programming of the thermal systems should provide the capability for cyclic fatigue testing of first-wall materials and components. High-speed automatic data recording and on-line data analysis is desirable for more accurate correlation with analytic model predictions and the failure analysis. These facility requirements and desirable features are summarized below:

- Minimal Requirements
  - > 0.1-m^2 first-wall test area
  - > 10-MW/m^2 peak thermal loading
  - vacuum or low-pressure (few mtorr) hydrogen environment
  - temperature, heat-flux, stress-deflection instrumentation
  - liquid (H_2O) cooling system for first-wall test panels
  - use of existing facility.
- Desirable Characteristics
  - ~ 100-MW/m^2 thermal loading for transients
  - gas or liquid-metal cooling capability
  - on-line data reduction.

b. Review of Existing Facility. LASL offers a combination of existing facilities at one of its Technical Areas (TA-46) that provides for experimental heat-transfer, thermodynamics and fluid-flow studies for the thermohydraulic and thermomechanical first-wall test programs. The facilities, originally developed as part of the nuclear rocket program, have most recently been used to perform loss-of-coolant and loss-of-flow tests of gas-cooled fast-fission reactor components.

These facilities are described on Attachment #5 and include four test cells, a 40-ft-tall high-bay, a 2.5-MW (dc) power supply, a 500-kW dc power supply, a dual Hewlett Packard data acquisition system, pressure and vacuum vessels suitable for controlled environment tests, auxiliary buildings and equipment necessary for test support. The total power supply to the test area is about 8 MW. A modern digital data acquisition system exists at the facility. This system and its instruments presently include two Hewlett-Packard 3052A data acquisition systems comprising:

- Two HP 9825 calculators (64 kbytes of memory each)
- 210 channel capacity
- Two full width line printers
- A four-color plotter
- Three flexible disk drives (0.5 Mbytes per disk)
- Two six-digit-resolution voltmeters
- One HP 5992 gas chromatograph/mass spectrometer
- A laser doppler velocimeter
- A hot-wire anemometer
- Two 75 channel chromel alumel thermocouple reference junctions.

Referring to Attachment #5, this test cell is located at Technical Area 46 (TA-46). The TA-46 test-cell complex comprises two main buildings connected by an underground tunnel. One building houses the 2.5-MW dc power supply, while the second building contains four test cells, a high-bay addition, and three control rooms. A connecting tunnel contains busbars that connect by means of an automated dc-power distribution system the test cells to the 2.5-MW dc power supply. Office space and a test preparation area are located adjacent to this facility, as is indicated in Attachment #5, which provides a complete list of the site buildings.

The test-cell/control-room building has a nominal width of 20 ft, and provides cells of varying sizes. Test Cell #1 (15′ × 40′) is the largest of the available cells and has an opening directly into the high-bay addition. Test Cell #2 (20′ × 15′) has direct visual access to Control Room #2 (20′ × 42′). The remaining control rooms (#1-20′ × 22′ and #3-20′ × 27′, respectively) also have visual contact with their adjoining test cells.

Each of the four test cells has two parallel reinforced concrete walls (10" thick) with minimum ceiling heights of 10 ft. The high-bay addition (15′7" × 12′4") adjoining Test Cell #1 is a 40-ft-tall structure with catwalks at 10-ft increments. This bay has a removable roof, a 10 ton overhead crane a half-ton crane and 18-ft-tall doors that allow for easy access to large equipment and experimental setups.

All test cells are served with natural gas, water, compressed air, and ac electric power. A high-pressure gas distribution system can simultaneously supply three different gases to each test cell from tube trailers through two 1-in. stainless-steel-lined mains (25 MPa capacity) and a 1-in. stainless-steel service drop (16.5 MPa). Recirculating cooling water is distributed to the test cells by 8" mains over the building length.

The dc power supply consists of two 1.4-MW synchronous motors, each driving two 0.625-MW dc generators. The generators are each rated at 75 V and 8333 A. The generators can be operated either in parallel or in series combinations, providing 75 V, 150 V, and 300 V at 33.3 kA, 16.7 kA, and 8.3 kA, respectively. A 12-kA circuit breaker protects each leg of each generator circuit, resulting in a total of eight breakers. One of the motor/generator sets has been modified to permit the series operation of its generators at a combined potential of 175 V.

Power is supplied to the cells through an underground tunnel extending the length of the test-cell building. Aluminum busbars, cooled by natural convection and with a maximum continuous duty rating of 16 kA, distribute power _via_ copper risers in each of four test cells. Water-cooled power cables, rated at 20-kA capacity, provide power to the high-bay addition.

A separate 500-kW dc-power supply, rated at 2000 A and 250 V, is housed in the area. This generator distributes power directly to the high-bay addition by a protected copper cabling system. Two portable dc-power supplies, rated at 150 A and 64 V, are also available.

Central control of the 2.5-MW dc supply is located in Control Room #2, but remote control can be delegated to any of the three control rooms. Remote programming for all of the existing power supplies can be accomplished through the Hewlett-Packard data controller in Test Cell #2.

Experimental instrumentation is centrally controlled by a dual Hewlett-Packard data acquisition system in Test Cell #2. Two HP-9825 calculators (64 kbyte memory each) control data acquisition and provide for the remote regulation of power supplies and experimental devices in the four test cells. The data system includes two voltmeters with 6-1/2 digit resolution and data scanning rates of approximately 20 readings per second, and a third voltmeter capable of handling about 4500 burst readings per second on one channel with 3-1/2 digit resolution. All data collected can be stored on any of three flexible-drive disk systems. Two printers, a plotter and strip chart recorders provide on-line information display.

Measurements of temperature, pressure, mass flow, and power can be made with existing instrumentation at various accuracy levels. Highly specialized instruments, such as a 15-mW He-Ne laser doppler velocimetry system, a temperature-compensated hot-wire anemometry system, and a gas chromatograph/mass spectrometer complement the extensive, basic instrumentation available at the facility.

The site provides a number of existing vacuum and pressure vessels that can be used in the proposed test program. The principle chamber planned for use in TPE-I(Phase I) is also shown in Attachment #5. This vacuum chamber provides a useable test space of approximately 2-m in diameter and 2.5-m in length, resulting in a useable volume of $\sim 8\ m^3$. Full-diameter access is available for test panel installation. Four ports are provided for test observation, instrumentation or other purposes. This chamber is fitted with a high-speed pumping system including mechanical roughing pumps, turbomolecular pumps and sublimation pumps. Electrical power and coolant feedthroughs are available in a fixed end-flange. The main shell of the chamber, excluding the end-flange holding the pumping system and feedthroughs, can be moved on rails to permit unrestricted access to the first-wall test.

Other test chambers are available at the site and may be adapted as required for use in TPE-I experiments. The largest chamber available is located in the high-bay addition and is 192" tall by 30.6" diameter. This vessel has numerous ports suitable for instrumentation as well as gas and water flow.

A shop, that is complete with lathe, mill, band saw, cutoff saws, drill press, ultra-sonic cleaners, and oxy-acetelene, TIG and other arc-welding equipment is available for test assembly preparation in Building WA-77. (Attachment #5). A remotely programmable, cylindrical grinder suitable for contouring graphite heater rods is situated in Test Cell #3. The site also supports a small machine shop, a small office, a lavatory and locker room, and an equipment room that houses the main air compressor for the entire facility. The test facilities were established for heat transfer, thermodynamic and fluid-flow experimental work and, therefore, will require minimal modification for use in the TPE-I test program.

c. Facility Modifications. No fundamental modification of the existing test facilities will be necessary for the TPE-I(Phase I) test program. The addition of radiant heater panels to the vacuum enclosure described in Sec. II.C.3.c. will enable testing of 0.1-m^2 samples at thermal flux levels of $1\text{-}2\ \mathrm{MW/m^2}$ early in the program. The extensive instrumentation and data reduction capability in existence and operational at the test site will also serve to expedite the first tests. Use of this existing facility will permit the conduct of basic first-wall tests well within the first year of the program while the parallel development of additional capability is proceeding. Modifications considered for development will include the addition of a high-energy-density heat-flux source for transient testing at localized thermal-flux levels up to $100\ \mathrm{MW/m^2}$. The method for implementation of these pulsed-power levels will be determined by initial (Phase 0) program studies. Candidate sources for evaluation will include electron-beam or inductive (Joule) heating generators.

The existence of four separate test cells with shared instrumentation and control, combined with the electrical power supplies and liquid- and gas-coolant distribution systems at the proposed test facility, provide the basis for considerable test-program growth at a common site.

4. Analytic Model Development. Based on an initial, conceptual definition of each FW/B/S test item, a preliminary stress and thermal analysis will be performed using a simple finite-element model for resolution of configuration details. This analysis will serve to identify areas of design requiring major modification or improvement. Once the final test configuration

is defined, a detailed three-dimensional, thermomechanical finite-element model will be developed. If required for specific materials or material interfaces new code features will be developed using as a basis one of the presently operating codes (e.g., ADINA).

The three-dimensional model will be used to determine the kind and location of instrumentation for the tests based on predicted temperature-time history, stresses, and deflections. These initial predictions will be verified by data correlation from early tests, and adjustments to both the model and the test set-up will be made as necessary. This correlation of predictive analysis with test data will be a major objective of the test program. Once optimum instrument locations are established, test parameters will be varied with the major goal being the correlation and upgrading of the model and the reviewing and revising of the test configuration after each operation. Long-term analytic development needs will be determined and special emphasis placed on obtaining proper data to benchmark these developments.

5. Instrumentation Development. A wide range of transducers is available to monitor both steady-state and transient responses. These temperature, stress and deflection transducers will be used in conjunction with a fully-automatic data acquisition system to give a time/space resolved data array for a given first-wall configuration that in turn can be compared with an appropriate computer simulation(s).

a. Transducer Selection. Development of instrumentation for the TPE-I test program will be directed towards three objectives. The first objective is the acquisition of engineering data from the tests for use in material and design evaluation. A second and equally important goal is the verification of analytic predictions of temperatures, stresses, and heat flow in order to develop confidence in the analytic tools to be used in ETF/FED and prototype fusion reactor designs. A third objective is the development of instrumentation techniques capable of being used for facility monitoring and diagnostics in a prototype facility. This latter purpose necessitates the consideration of environmental effects which may not be present in the actual TPE-I tests, (e.g., high neutron fluences and magnetic fields). For this reason the following discussion of instrumentation considerations for the TPE-I test program concerns itself with prototypical environments.

The physical quantities of primary interest in the program will be temperature, pressure, strain, displacement, velocity, and acceleration. In addition flow parameters for the cooling system must be measured. As the phenomena to be measured generally will be transient and will be recorded remotely, transducers that convert the physical quantity of interest to an electrical signal will be necessary. Numerous electrical transducers are available for the measurement of each of the physical parameters listed. Because of the severe environments encountered in this usage, however, selection of appropriate transducers will be critical for longer-term applications.

It is anticipated that preliminary testing of the selected transducers will be required. These instrumentation tests will not constitute transducer development; rather, they will consist of calibration of performance in the use environment and evaluation of mounting and protective modification.

b. Candidate Instrumentation. Variable-resistance (ΔR) type temperature transducers can be used for accurate measurement up to 1100 K (e.g., Trans-sonic, Inc.). These temperature transducers are made of platinum or nickle alloys and demonstrate good stability in a radiation environment. When used at the higher operating temperature, installation is usually made by welding.

Several kinds of thermocouples are available for use in the 800-700 K temperature range (e.g., 60% iridium-40% rhodium/iridium; platinum-10% rhodium/platinum, from Hy-Cal Engineering). Thermocouples are used widely in a radiation environment, resulting in the availability of a large amount of radiation-effects data.

Pressure, strain and acceleration transducers are normally embodied in the following three kinds:

- variable resistance (metal)
- variable resistance (semi-conductor)
- piezoelectric transducers.

The ΔR semi-conductor transducers generally show considerable distortion at high temperatures and radiation fluence. Consequently, if the measuring techniques developed in the course of TPE-I are to be extended to more realistic conditions in the future, semi-conductor ΔR transducers should be avoided. The metal ΔR transducers have been used to measure static strains (or pressures) at the temperatures as high as 800 K. Temperature compensation and transducer mounting present the major problems. At temperatures envisaged for the TPE-I tests, strain gages are often attached by welding (e.g., Microdot, Inc., Weldable-wire strain gage). Radiation effects on metal ΔR transducers have been studied extensively; generally the zero-calibration shifts under continuous irradiation, and, therefore, long-term strain or pressure measurements are difficult.

Although not as widely used as the ΔR transducer the variable-capacitance (ΔC) transducer for measurement of motion, strain and pressure has unique advantages when a high-temperature measurement is required. These devices are also operable in a radiation environment.

Piezoelectric transducers, as applied to the measurement of pressure or acceleration, have been specifically designed for high-temperature in a radiation environment (e.g., Endevco and PCB Piezotronics provide devices for use in PWR and LWR systems). Transient radiation fields produce false signals, but these signals generally are small compared to the accelerations or pressure-induced signals. The piezoelectric transducers should be used in the charge mode under these circumstances, since the field-effect transistors used in a voltage mode are damaged by radiation.

6. TEST OPERATION

a. Steady-State Operation. Initial tests will be performed on panel-coil and tube-sheet first-wall configurations that are cooled with pressurized water and operated in a steady state. Radiant heating *in vacuo* or low-pressure (few mtorr) hydrogen will be used on an array of first-wall element sizes to permit heat-fluxes in the range 1-10 MW/m^2 (steady-state) to be examined. All test arrays will be representative of a fully-engineered first wall, supporting appropriate sizes and numbers of weldments, bends, asymmetrics, and mechanical joints. The stainless-steel, prime-condidate alloy (PCASS) will be used with and without low-Z coatings. Similarly, smaller samples of first-wall components (i.e., limiter edges, protective armor) will be examined under steady-state heat fluxes in the range 10-100 MW/m^2. In addition to mapping temperatures, stresses and deflections for a range of coolant conditions, non-symmetric heating and support configurations, system responses as reflected by changing material properties, mass transport, changing heat-transfer characteristics (e.g., fouling) will be documented under conditions of long-term testing for both the generic first-wall and first-wall component configurations. For selected test elements, state-of-the-art post-test diagnostics will be performed on critical first-wall regions and first-wall components.

Since it is desirable to carry any test sequence to a failure mode, and since the increased heat loads required to induce failures can be achieved only by reduced test-element size, considerable care must be used in relating any observed failure mode to either higher heat loading *per se* or to geometric and coolant-configurational changes that may result from a need to reduce test element size. The TPE-I program will rely strongly on both engineering judgment, state-of-the-art analysis and post-test diagnostics to separate these effects as causes for specific failure modes observed under high-heat-flux, steady-state conditions.

b. Transient Effects. A large number of transient heat-flux conditions can be envisaged and simulated by the TPE-I experimental program. For example:

- normal cold startup to a steady-state operating point
- oscillations in the steady-state heat flux that are small relative to the steady-state value
- localized transients wherein the average heat flux is constant but time variations occur across the bulk of the test element
- single-shot intense heat dumps that far exceed steady-state design values either over the entire test element or localized to a small portion of the test element
- sudden changes in the coolant flow or geometry.

The TPE-I test facility and general approach proposed herein is capable of simulating all of the above conditions through: a) control of the primary radiative heating source (e.g., power control or control through the use of a variable shutter or focusing mechanism; b) use of auxiliary, intense heaters (e.g., electron beams, inductive or dc resistance heaters, neutral or ion beams); or c) combination of the above. Failure of a first-wall test element is expected to occur either singularly (e.g., by melting or loss of coating) or over a period of time through fatigue (primarily at bends, weldments and other areas of structural imperfection). Since these transient tests to failure generally are expected to be time intensive and difficult to interpret, this phase of TPE-I should be discussed with both ANL, the ETF/FED design team and the fusion technology community as a whole. It is nevertheless recommended that the following three categories be used in classifying any series of transient TPE-I tests:

- "steady-state" oscillations and recurring "local" transients
- diminution or loss of cooling capacity
- intense, local single-shot heat dumps (i.e., plasma disruptions, leading to intense surface and bulk power densities).

7. Test Data Reduction, Evaluation and Dissemination. Data acquisition and reduction throughout the test program will be directed towards the following objectives: a) correlating analytic predictions; b) providing a basis for component evaluation; and c) developing an engineering data base for facility design. It is assumed that the data requirements for these three objectives will be similar. Each will require the ability to record large quantities of data at medium rates (i.e., at a maximum one data point per 1-10 ms). In the case of the analytic data correlation program, capability for on-line data reduction and display will be used to minimize turn-around time. The basis for the test data handling system will be the Hewlett-Packard system now in existence at the proposed test facility (Sec. II.C.3.b.) This programmable data manipulation capability permits on-line display of reduced data by plotter, printer or CRT. Incoming test signals are digitized and stored in the mini-computer memories up to a total of 120 kbytes. When the storage capacity of the minicomputers is reached the recording is interrupted briefly and the accumulated data dumped to a floppy disk. The disk provides a storage capacity up to 1.5 Mbytes, allowing continuous monitoring of long-term tests. The storage capacity of the minicomputer memories is sufficient to record data through initial transients, thereby minimizing data loss in the transfer process. For post-test data reduction the disk stored data may be transferred back through the processor for reduction and display. The data handling capability of the existing system provides the means for rapid reduction of large quantities of test information to an appropriate form for dissimenation and use in the engineering design of fusion reactors. Format and content of the engineering design data documentation will be established in the Phase 0 effort.

D. INVOLVEMENT OF SUPPORTING PARTICIPANTS

As seen from the Laboratory organization proposed for TPE-I (Attachment #1) all aspects of this task are adequately covered by an appropriate level and range of expertise. The primarly fusion-relevant input and expertise will be channeled through Group CTR-12, which through the program manager will serve as the interface between the LASL TPE-I activity, ANL, and other TPE activities. If it appears appropriate and if available experts are identified, however, they will be incorporated into the program as required.

E. PHASE I SCHEDULE

A preliminary schedule of Phase I activities is given in Attachment #3. This schedule is based on the general approach suggested in Sec. II.A and correspondingly will be subject to refinement as the Phase 0 proceeds. Generally, first tests on existing equipment using a large but relatively simple panel-coil or tube-bank first-wall segments should begin under full instrumentation by May 1981 with the first conclusive results emerging for review from Phase I in the June-July 1981 time frame.

F. PRINCIPAL AND KEY INVESTIGATORS

A general LASL organization proposed for the execution of the TPE-I task is given in Attachment #1. Given below is a breakdown of key individuals who would contribute to the TPE-I(Phase I) effort. Resumes for each key investigator are found in Attachment #6.

- Principal investigator for management of Phase I
  M. A. Merrigan 1/4 time
- Co-principal investigator for fusion technology liaison
  R. A. Krakowski 1/4 time
- Experimental program investigator
  M. A. Merrigan 1/2 time
- Analytic program leaders
  T. A. Butler/G. E. Cort 1 time
- Test evaluations leader
  R. C. Dove 1/4 time
- Test facility designer
  G. E. Cort 1/4 time

Total 2-1/2 man-years

G. TECHNICAL AND MANAGERIAL CAPABILITIES

The technical and managerial organization (Attachment #1) that will be used to implement TPE-I reflects an established and well-known capability. Los Alamos Group Q-13 is well versed in the experimental and analytical methods needed to fulfill all TPE-I(Phase I) tasks. In addition, Group Q-13 will be supported by experts in other areas of fusion technology (Group CTR-12), advanced engineering analysis/design (Group WX-4) and materials technology

(Groups CMB-3, -5, and -6). Coordination of this effort and direct communication with ANL will be assured through a program manager whose overall responsibilities include all Laboratory-wide efforts in magnetic fusion development/technology.

H. UNIQUE FACILITIES AND OTHER RESOURCES

The facilities that will be applied to TPE-I(Phase I) have been described in Sec.II.C.3. and Attachment #5. These facilities will be shared by other programs to an extent that is yet to be determined. The existence of these facilities, particularly the power supplies, test bays, instrumentation and support shops, represents a considerable savings of both cost and effort for the TPE-I(Phase I) program. The cost burden to a TPE-I(Phase I) program that uses these facilities and modifications thereof has not yet been determined, but should be minimal; generally, it is expected that a major part of such costs will be related only to maintenance, repair, and retrofitting.

I. DATA RELEASE STATEMENT

The Los Alamos National Laboratory is a national laboratory and, as such, all unclassified information generated by the laboratory is and will be made available to the public.

J. COST INFORMATION

At the 2.5 man-year level-of-effort (Sec.II.F.) projected for TPE-I(Phase I) and an annual staff-member cost of 96 k$/y, the resulting 240 k$ plus 40 k$ for materials and supplies would result in a target cost of 280 k$ for FY 1981. The 96 k$ per staff member year includes a 70.5% overhead charge on salary plus fringe benefits. These costs are typical of other magnetic fusion energy programs funded directly by the Office of Fusion Energy/DOE at Los Alamos.

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ATTACHMENT #1. LOS ALAMOS ORGANIZATION FOR THE IMPLEMENTATION OF TPE-I.

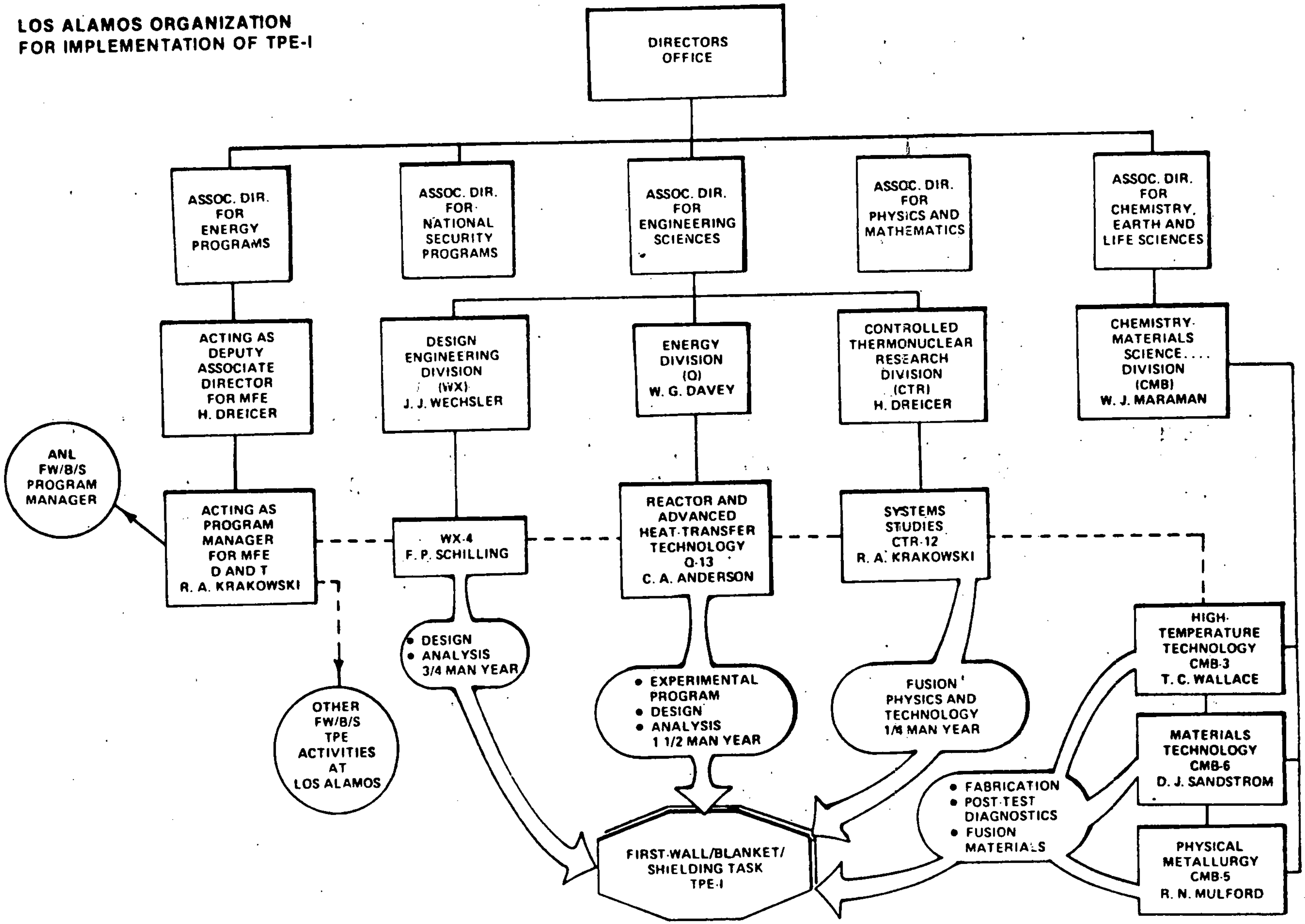
ATTACHMENT #2. APPROACH TO TPE-I(PHASE I).

ATTACHMENT #3. TPE-I SCHEDULE

ATTACHMENT #4. PARTIAL LIST OF COMPUTER PROGRAMS IN USE AT LOS ALAMOS WITH APPLICABILITY TO TPE-I.

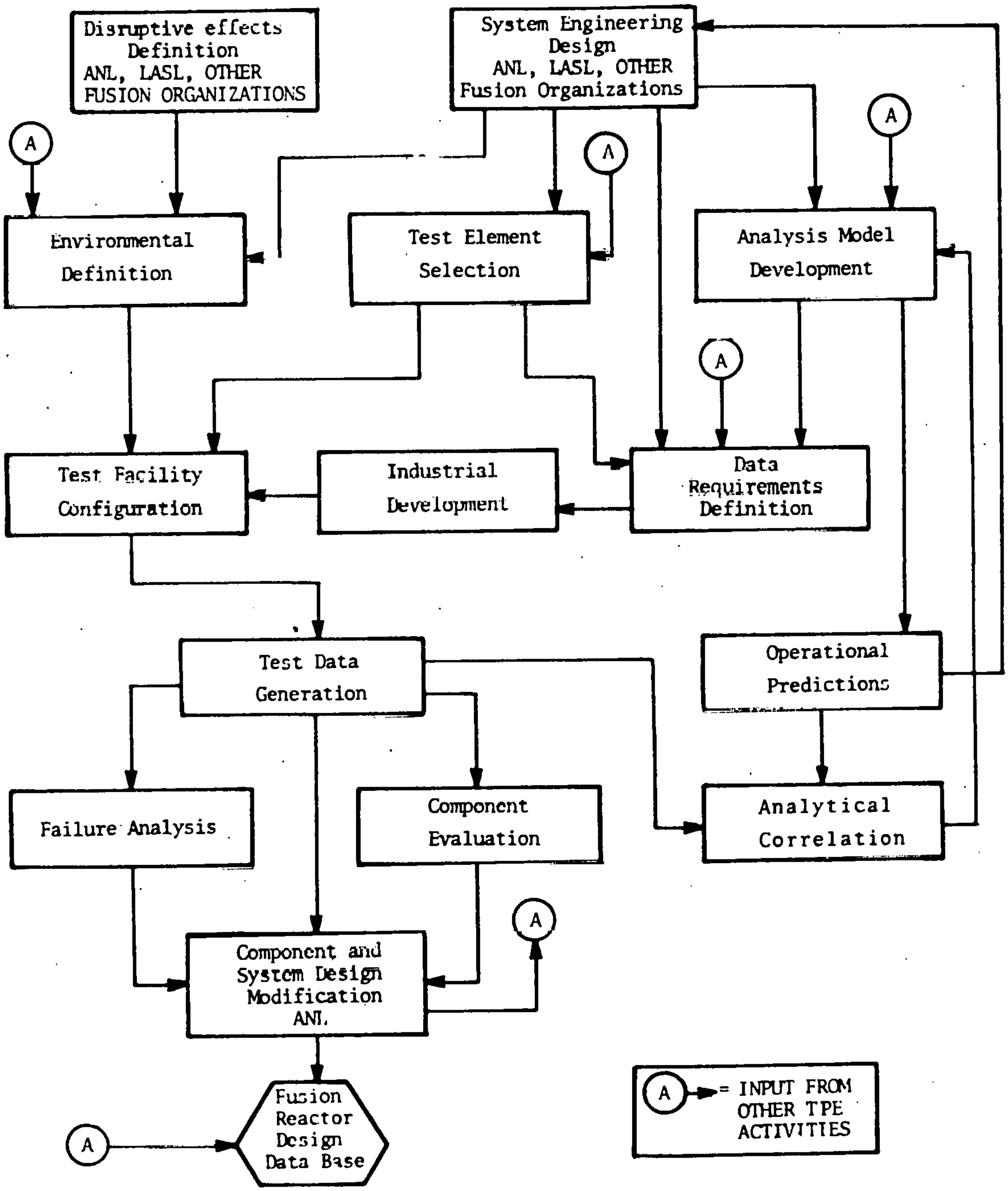
ATTACHMENT #5. GENERAL DESCRIPTION OF EXISTING TEST FACILITY TO BE USED FOR TPE-I.

ATTACHMENT #6. RESUMES OF KEY INVESTIGATORS


LOS ALAMOS ORGANIZATION
FOR IMPLEMENTATION OF TPE-I
DIRECTORS OFFICE
ASSOC. DIR. FOR ENERGY PROGRAMS
ASSOC. DIR. FOR NATIONAL SECURITY PROGRAMS
ASSOC. DIR. FOR ENGINEERING SCIENCES
ASSOC. DIR. FOR PHYSICS AND MATHEMATICS
ASSOC. DIR. FOR CHEMISTRY, EARTH AND LIFE SCIENCES
ACTING AS DEPUTY ASSOCIATE DIRECTOR FOR MFE H. DREICER
DESIGN ENGINEERING DIVISION (WX) J. J. WECHSLER
ENERGY DIVISION (Q) W. G. DAVEY
CONTROLLED THERMONUCLEAR RESEARCH DIVISION (CTR) H. DREICER
CHEMISTRY-MATERIALS SCIENCE DIVISION (CMB) W. J. MARAMAN
ANL FW/B/S PROGRAM MANAGER
ACTING AS PROGRAM MANAGER FOR MFE D AND T R. A. KRAKOWSKI
WX-4 F. P. SCHILLING
REACTOR AND ADVANCED HEAT-TRANSFER TECHNOLOGY Q-13 C. A. ANDERSON
SYSTEMS STUDIES CTR-12 R. A. KRAKOWSKI
HIGH-TEMPERATURE TECHNOLOGY CMB-3 T. C. WALLACE
MATERIALS TECHNOLOGY CMB-6 D. J. SANDSTROM
PHYSICAL METALLURGY CMB-5 R. N. MULFORD
• DESIGN • ANALYSIS 3/4 MAN YEAR
• EXPERIMENTAL PROGRAM • DESIGN • ANALYSIS 1 1/2 MAN YEAR
FUSION PHYSICS AND TECHNOLOGY 1/4 MAN YEAR
• FABRICATION • POST-TEST DIAGNOSTICS • FUSION MATERIALS
OTHER FW/B/S TPE ACTIVITIES AT LOS ALAMOS
FIRST-WALL/BLANKET/SHIELDING TASK TPE-I

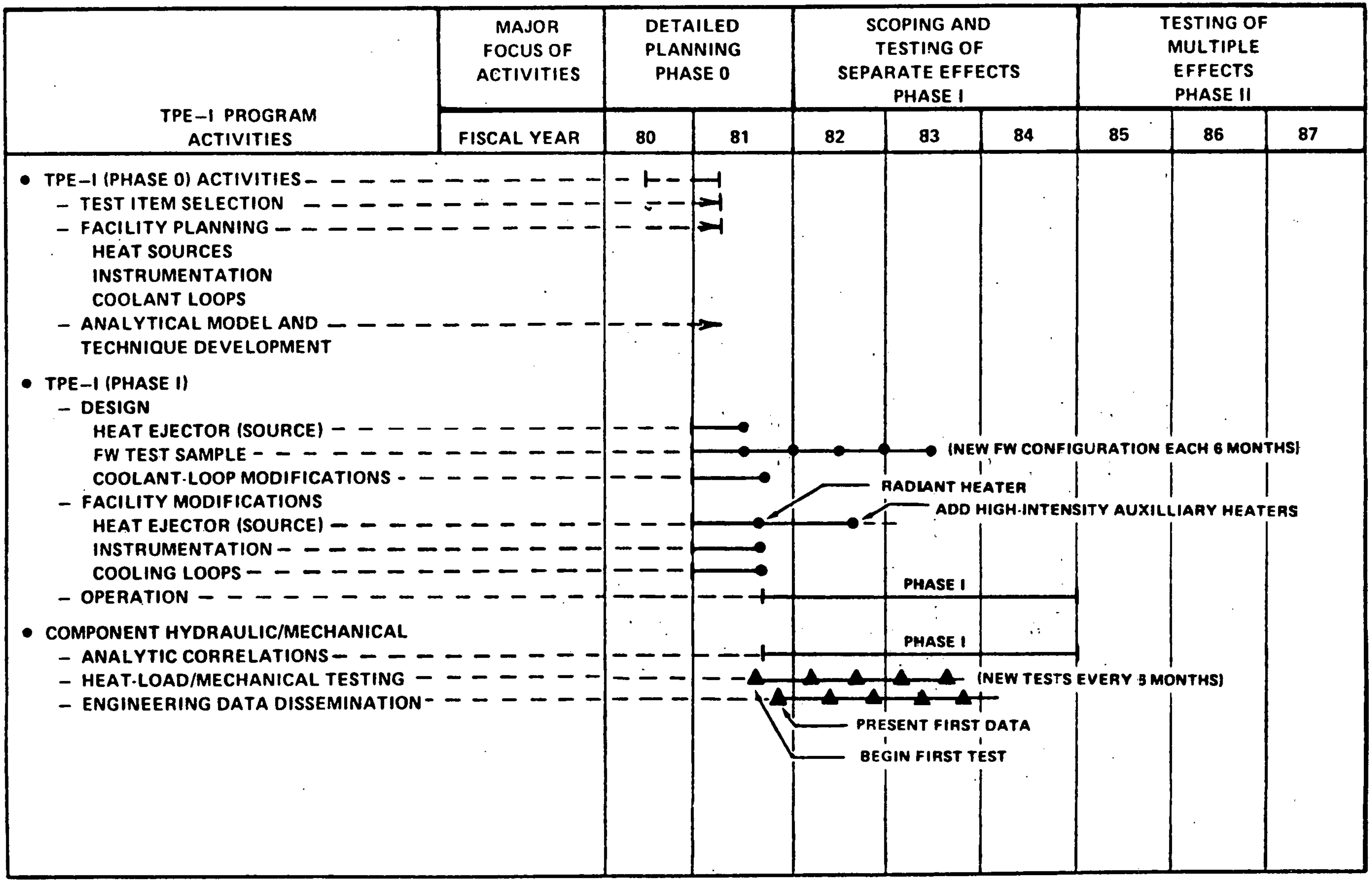
Attachment #2

APPROACH TO FIRST-WALL AND
FIRST-WALL COMPONENT TEST PROGRAM
(T P E I)



Attachment #3

TPE–I: FIRST-WALL COMPONENT THERMAL-HYDRAULIC/THERMOMECHANICAL TESTING: FOCUS, SCHEDULE AND MAJOR MILESTONES FOR PHASE I ACTIVITIES



ATTACHMENT #4

COMPUTER PROGRAMS IN USE AT LOS ALAMOS WITH DIRECT APPLICATION TO THE FW/B/S PROBLEM
(Partial list)

Structural Analysis (all use the finite-element method)

SAPV - Static and dynamic analysis of linear, three-dimensional systems.

TSASS - Thermal and stress analysis of static and dynamic, linear and nonlinear, three-dimensional systems.

ADINA - Static and dynamic analysis of linear and nonlinear three-dimensional systems.

SPAR - Interactive stress analysis of static and dynamic, linear, two-dimensional systems.

Heat Transfer*

MITAS - Interactive finite-difference solver for large multidimensional lumped-parameter, i.e., resistor capacitor (R-C) network representations of thermal systems.

AYER - Versatile finite-element program to solve the transient, nonlinear, two-dimensional heat-conduction equation, including effects of anisotropy and material movement.

ADINAT - General nonlinear finite-element solution to three-dimensional, transient heat-conduction problems.

Fluid Flow

VNAP - Versatile finite-difference program for calculating viscous, two-dimensional, steady and unsteady, compressible internal flow. Typical problems that can be solved are flow in pipes and ducts, subsonic and supersonic inlets, free-jet expansions, and nozzles.

SOLA - Versatile three-dimensional transient in compressible viscous flow with variable mesh and free surface.

Chemical Reaction

CHEMSHARE - A broad-based process simulator with capability for developing process flow sheets, heat and material balance data, physical property data, phase equilibria, and equipment sizing and design.

GENMIX - General code for predicting steady, two-dimensional fluid flow with heat and mass transfer and chemical reactions.

2DE - Reactive, multicomponent, two-dimensional Eulerian hydrodynamic code used to model shock waves, detonations, and initiation of detonations in homogeneous and heterogeneous explosives.

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*Because of the analogy between various field variables, these codes have been used to solve other field problems, such as flow in a porous medium.

ATTACHMENT #5

General Description of Existing Test Facility to be used for TPE-I.

Figure Captions for Attachment #5.

Fig. 1. TA-46 facilities at Los Alamos applicable to FW/B/S test program.

Fig. 2. 2.5-MW dc generator set provides a programmable source for radiant heater power.

Fig. 3. Central data acquisition and control station employs two HP-9825 data processors.

Fig. 4. Vacuum chamber to be used for FW/B/S test program.

Fig. 5. Schematic diagram of experimental apparatus proposed for FW/B/S test program.

TABLE I

BUILDING SUMMARY

LASL Structure Designation	Purpose
WA-16	Test cells, control rooms, 2.5 MW dc power supply and distribution system
WA-17	Lavatory, air compressor, location for additional power supply (has access to tunnel)
WA-18	dc power distribution tunnel
WA-19	1500 KVA transformer - input power to Bldg. 17
WA-37	High pressure compressor and gas handling equipment, including tube-trailer hookup facilities
WA-58	Support shop facility, lavatory, locker room, equipment room, office
WA-77	Test article preparation and staging area
WA-86	Cooling tower
WA-87	Cooling water treatment and circulation equipment, heat exchanger
WA-105	5000 KVA Transformer - ac power supply for 2.5 MW MG set
ULR-385	Field office with lavatory

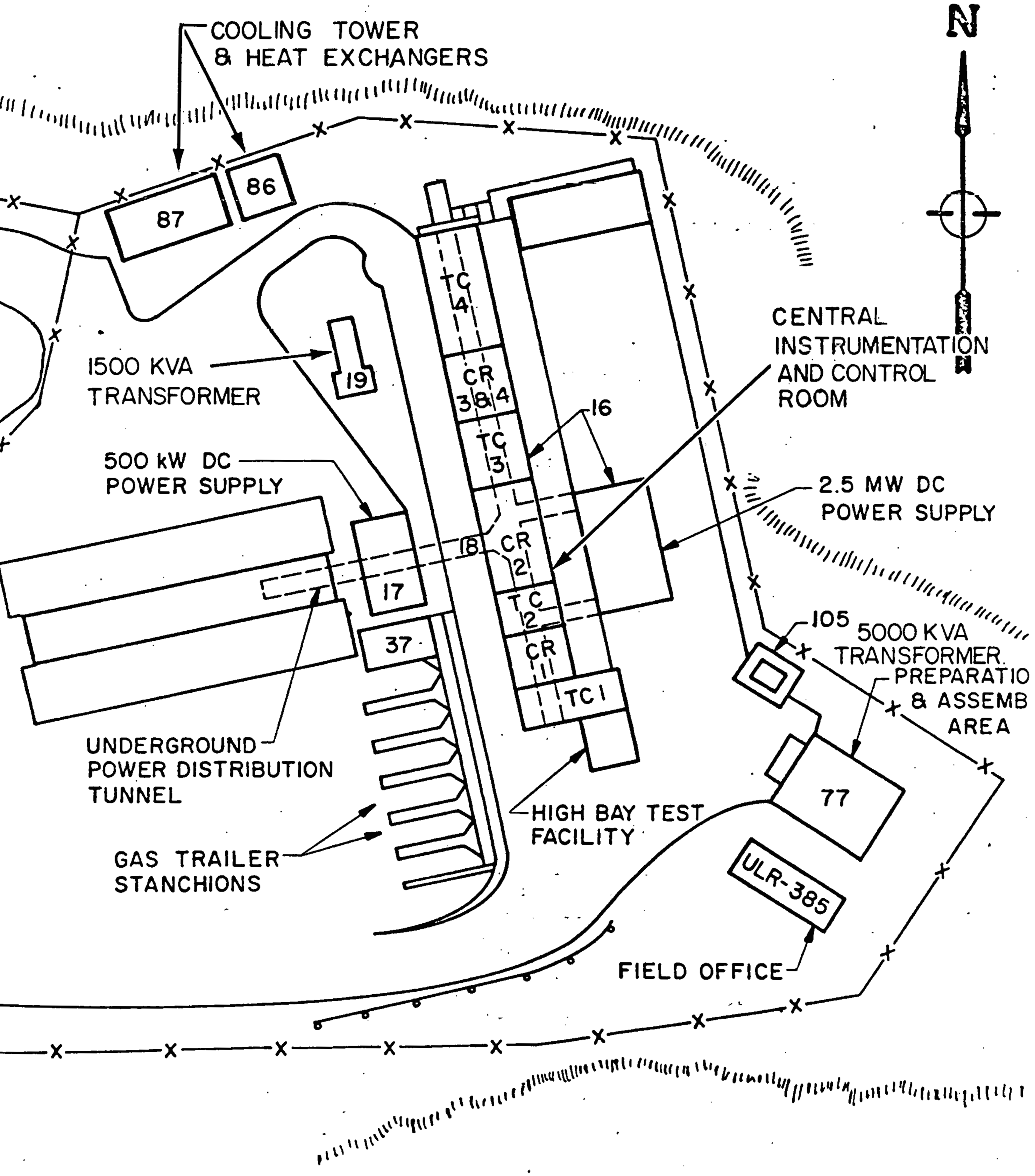
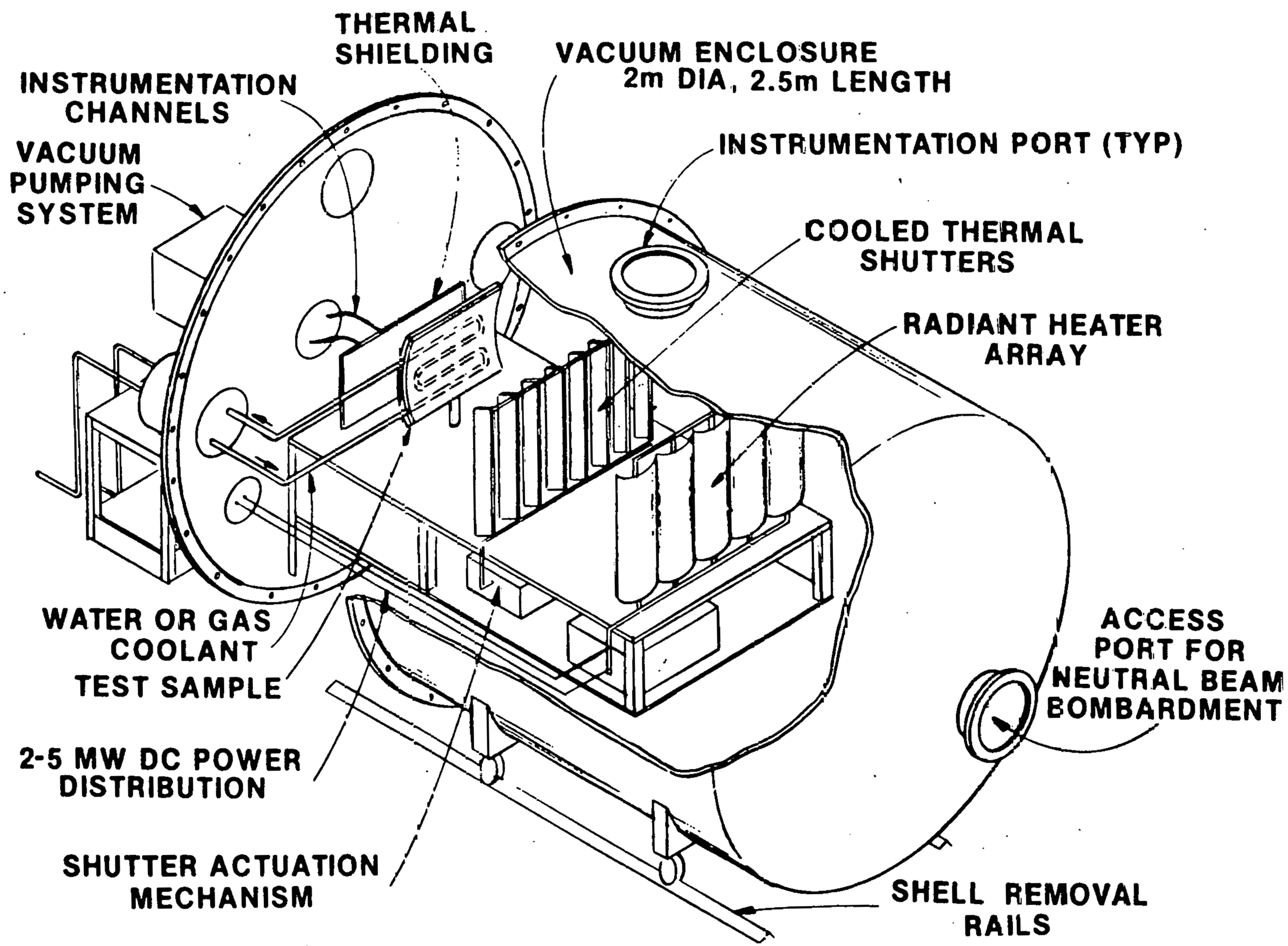


Fig. 1. TA-46 FACILITIES APPLICABLE TO THE FW/B/S TEST PROGRAM



FIRST WALL THERMAL-HYDRAULIC TEST FACILITY

Fig. 5.

ATTACHMENT #6

Resumés of Key Investigators

QUALIFICATIONS AND ASSIGNMENT OF PROGRAM PERSONNEL

M. A. Merrigan

Education BSME, MSME, University of Southern California

Experience: 2 years at Los Alamos National Laboratory
22 years total

Specialities: Thermodynamics, Fluid Mechanics, Heat Transfer, Chemical Kinetics

Present assignment is project leader for the Ceramic Heat Pipe Program in experimental DOE fossil fuels program conducted through the Pittsburg Energy Technology Center (PETC). Activities in this capacity consist of the technical management of LASL and subcontract investigations, preparation of reports and reviews, and the development and negotiation of budgets.

Immediate prior position was at Highes Aircraft Company as Head of the Mechanical Design and Analysis Section in the Communications and Radar Division from 1974 to 1979. Section responsibilities consisted of the physical design and implementation of radar and communications systems including direction of in-house and subcontract fabrication of electronic devices and subsystems ranging from printed circuit and stripline boards through ferrite and diode phase shifters to servo-control equipment. Duties included technical supervision, cost and schedule estimation and control, vendor selection and monitoring, and preparation and presentation of technical proposals and reviews. This position entailed the direction of approximately 50 analysis and design personnel.

Previous assignment at Hughes was as Head of the Advanced Development and Analysis Section for the Technical Services Group Office from 1968 to 1974. In this capacity was responsible for technical support to the line engineering electronic organizations for thermal analysis, dynamics, and design for military field environments and tests. Responsibilities as Section Head included the development and direction of in-house support effort as well as engineering R&D programs encompassing program definition, proposal preparation, negotiation with sponsors, and supervision of technical efforts. Programs developed during tenure included an investigation of microelectronics packaging techniques conducted for USAF; thick-film hybrid microelectronic thermal design models for use in circuit layout, and the application of ebullient cooling techniques to high voltage power supplies. The application of advanced thermal control techniques to military electronics equipment through the utilization of heat pipes and phase change materials for energy storage was investigated under the sponsorship of USAF, Wright Patterson Air Force Base; NASA, Huntsville; and U.S. Army Electronics Command. Other programs involved the feasibility analysis of combined environment test facilities for the U.S. Navy and the evaluation of dynamic test methods for high intensity shock simulation. While assigned to the Technical Services Group Office also served as Chairman of the Technical Papers Review Committee

Mr. Merrigan has conducted feasibility studies of space environmental facilities for propulsion research; experimental investigations of hypergolic propellant reactions in low pressure, cryogenic environments; analysis of SNAP-8 components.

Publications

Technical Report No. 23-34, "Design Guide for Pressurization System Evaluation Liquid Propulsion Rocket Engines," Aerojet General Corporation, September 1962.

Technical Report No. USC-86-101, "Investigation of Chemical Non-Equilibrium Effects in Viscous Shock Layers," USC Engineering Center, May 1963.

Technical Report No. AFRPL-TR-65-19, "Propellant Reaction Characteristics in Space Environmental Test Facilities," Celesco Corporation, 15 January 1965.

"On the Spontaneous Ignition of Hypergolic Propellant Systems at Low Pressures and Temperatures," Paper No. 64-29, Western States Combustion Institute Meeting, Salt Lake City, Utah.

"Propellant Reaction Characteristics in Space Environment Test Facilities," AFRPL-TR-65-19, 15 Jan. 1965.

"Thermal Control of Airborne Electronic Equipment," AFFDL-TR-73-12, June 1973.

"Investigation of Novel Heat Removal Techniques for Power Transistors," ECOM-0021-I R&D Report October 1972.

"Pipelines as Instrumented Networks for Earthquake Prediction," NACE paper No. 59, Corrosion 1974 Symposium, March 1974, Chicago, Illinois.

"High Temperature Heat Pipes for Waste Heat Recovery," M. A. Merrigan and E. S. Keddy; AIAA 15th Thermophysics Conference July 14-16, 1980, Snowmass, Colorado; LA-UR-79-3437.

"Economics of High Temperature Recuperation Using Ceramic Heat Pipes"; M. Merrigan, American Society for Metals Conference, June 4, 1980, Pillsburgh, Pa.; LA-UR-80-1550.

"A Heat Pipe Heat Exchanger Model for High Temperature Recuperators," M. Merrigan, CUBE Symposium, Otb. 22-24, 1980, Lawrence Livermore National Laboratory; LA-UR-80-1985.

Activities and Honors: Tau Beta Pi, Pi Tau Sigma, American Society of Metals

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Education: B.Ch.E./M.S. The Ohio State University in Chemical Engineering (1961) Thesis: Gamma-ray Induced Chemical Reactions.
Ph.D. University of California, Berkeley, in Nuclear Engineering/Engineering Physics (1967). Dissertation: High-Temperature Interaction of Hydrogen Beams with Metallic Surfaces.

Present Position

Group Leader, LASL Group CTR-12
Magnetic Fusion Systems Studies

Honors

Tau Beta Pi
Phi Lambda Upsilon
Sigma Chi
AEC Fellowship (1961-62)
UOP Fellowship (1960-61)

Professional Experience

1962-65 - Research Assistant at Lawrence Radiation Laboratory, Berkeley site.
1965-66 - Acting Instructor of Nuclear Engineering at Berkeley
1967-68 - Research Fellow at Ispra (Italy) Center of Research for Euratom
1968-72 - Assistant Professor, Nuclear Engineering, OSU
1972-73 - Staff Member, Group N-5 (Thermionics Development), LASL
1973-present - Staff Member/Group Leader (1976), Group CTR-12 (Magnetic Fusion Systems Studies), LASL

Consulting Experience:

Battelle Memorial Institute (5/68-6/71).
Los Alamos Scientific Laboratory (6/70-9/70)
University of California, Berkeley (1976-1979)
University of Washington (1978)

Professional and Scientific Organizations:

American Nuclear Society
American Ceramic Society
American Chemical Society
Ohio Society of Professional Engineers and Surveyors
Registered Professional Engineer, State of Ohio

General Areas of Research Interest:

High temperature (inorganic) chemistry, solid-state physics, radiation effects in solids, physical chemistry of nuclear fuels, heterogeneous catalysis, chemical kinetics, adsorption phenomena, material creep, solid-state diffusion and transport phenomena, electrochemistry, corrosion (oxidation), Mössbauer spectroscopy, material and heat-transport problems associated with nuclear reactor safety, fusion technology and systems studies, plasma engineering.

RESEARCH EXPERIENCE AND RESPONSIBILITIES:

- <u>Research Experience at Los Alamos Scientific Laboratory</u>:
  (7/72-6/73; Supervisor: William A. Ranken)
  (6/73-6/77; Supervisor: Fred Ribe/Keith Thomassen)
  (6/77-present; Supervisor: Harry Dreicer)

As a staff member and group leader associated with fusion-reactor systems studies and plasma engineering, primary responsibilities have centered on the computational modeling and design of a wide range of magnetic confinement schemes for the production of fusion power. In the course of evaluating specific fusion concepts the following engineering and scientific topics were addressed: plasma physics, plasma engineering, electrotechnology and energy storage, thermomechanical and thermohydraulic design, systems design, tritium chemistry, radiation damage, engineering economics. In addition to performing system studies and technology assessments for a large number of fusion reactor concepts, a strong interaction was maintained in a range of areas of contemporary interest to and planning for the CTR Division.

As a staff member associated with thermionic reactor research, primary responsibilities were in the area of materials problems associated with fuel performance and fuel/cladding interactions. The major tasks in this effort which were actively being pursued at the termination of this program were:

- Development of a fueled irradiation capsule to resolve and enhance the long-term dimensional stability of thermionic emitters. The irradiation was scheduled for the Gulf General Atomic TRIGA reactor in San Diego.

- Development of a fueled irradiation capsule to resolve the short-term behavior of fission products and dimensional stability of thermionic emitters. The irradiation was scheduled in the LASL OWR.

- Development of a high volume percent (90 v/o) UO_2W cermet having high open porosity with good high-temperature stability. This high creep rate material is to form a fuel weakening region between the dense UO_2 core and the W clad and by stress relieving mechanism reduce the stress exerted by the fuel on the clad (emitter). Sintering and creep studies of the UO_2/W cermet were being performed, and fabrication procedures required to form this material into specific shapes for use in thermionic emitters were being resolved.

- The biaxial creep properties of columnar oriented UO_2 was to be studied out-of-pile.

- A means to enhance the creep properties of UO_2 via chemical additions were being researched.

- An out-of-pile test of a fueled thermionic emitter was in progress. This test was designed to simulate closely the thermal gradients expected in-pile. Fuel (UO_2) redeposition mechanisms and thermal ratcheting processes were to be studied. Additionally, the design bases selected for the two aforementioned irradiation experiments and a means to produce columnar oriented UO_2 for out-of-pile creep investigations were to be provided by the out-of-pile tests.

- Post-irradiation analyses of fueled capsule experiments and thermionic fuel element tests were being performed. This work involved fission gas retention studies, ^{235}U burn-up determination, metallography, and electron microprobe analyses. The goal of these analyses was to elucidate fission gas transport mechanisms, material compatibility, and certain chemical reaction zones.

- Development of analytical models to predict temperature distributions, fission gas transport and release, fuel redistribution, and stress field in fueled emitter was in progress.

- <u>Responsibilities at Ohio State University:</u>
  (9/68-7/72; Supervisor: Donald Glower)

<u>Course Responsibilities:</u>

- Mechanical Engineering 311 (511): <u>Heat Transfer and Fluid Flow</u>. Study of the fundamental principles of heat transfer and fluid flow. (Undergraduate)

- Nuclear Engineering 716: <u>Nuclear Safety</u>. Modeling theory developed and applied to nuclear systems to facilitate analysis of possible nuclear accidents, preparation of Preliminary Safety Analysis Reports, reactor siting criteria, reactor safeguard, with special emphasis on light-water power reactors. (Graduate/Undergraduate)

- Nuclear Engineering 743: <u>Nuclear Engineering I</u>. Experimental investigation of nuclear radiation interaction with matter, experimental verification of basic principles of atomic and nuclear physics, shielding, detector response, radiation detection systems. (Graduate)

- Nuclear Engineering 744: <u>Nuclear Engineering II</u>. Experimental nuclear reactor analysis, criticality experiments, measurement of nucleonic and reactor parameters. (Graduate)

- Nuclear Engineering 814: <u>Effects of Radiation Interactions in Matter</u>. Effects of high-energy radiation in gases, liquids, and solids with emphasis on basic interaction processes in solids, property change in reactor fuels and construction materials. (Graduate)

- Nuclear Engineering 862: <u>Breeder Reactor Design</u>. Breeding cycle in thermal and fast reactors, breeding rates and criticality calculations for breeding systems, fuel cycles, special design concepts, liquid-metal fast breeder technology, fast reactor physics. (Graduate)

- Research Responsibilities:

  - "Oxygen Concentration Cell Measurement of Ionic Transport Numbers in the Family of PZT Polycrystals," MS Thesis, Andre Ezis (1969).

  - "Defect Mechanisms in Lead Zirconate Titanate—AC Conductivity and EMF Measurements," Ph.D. Dissertation, John Burt (1970).

  - "Transport Phenomena at Liquid-Vapor Interface of Mercury Using a Radioactive Tracer," Ph.D. Dissertation, Kyral F. Wylie (1970). R. W. Brodkey, D. D. Glower, co-advisors.

  - "Pulse-Decay Phenomena in Lead Zirconate Titanate Ferroelectric Ceramics, MS Thesis, Ali Zied (1971).

  - "Effects of Hydrostatic Pressure on the Degree of Swelling of Uranium Monocarbide Induced by Fission Product Gases." This work represents an experimental in-pile study of fission gas swelling at high pressure (~ 10 kpsi) and high temperature (~ 1000-1500°C). The swelling rate will be measured in situ, and a range of stoichiometry ($UC_{1\ x}$) will be investigated. Ph.D. Dissertation, begun 1970, experimentation underway at Argonne National Laboratory.

  - "Study of the Oxidation of 304 Stainless Steel Using Mössbauer Spectroscopy." Thin (25 µm) foils of 304 stainless steel are oxidized in dry air in the temperature range 600-1000°C. Oxidation kinetics are resolved by continuous weight-gain measurements. Within selected oxidation regimes (nucleation, protective, and breakaway stages) the oxidized specimen is cooled and Mössbauer spectra are obtained. Phase changes are monitored and identified, both in the oxide phase (via oxide stripping) and in the metallic substrate. X-ray and electron microprobe analyses are used in conjunction with the Mössbauer spectroscopy to elucidate the oxidation processes. Ph.d. Dissertation, started 1970, work completed 1/73.

  - "Response of He/CH_4 Proportional Counter to Electron Irradiations in the Range 5-10 keV." The proportioning characteristics of He/CH_4-filled gas counters to low energy, electron irradiation are resolved. The limitations on energy resolution imposed by detector geometry, material of construction, gas composition, applied voltage, background radiation, and pulse-shaping circuitry are identified. System parameters of importance to energy resolution are optimized. MS Thesis, started 1971, completed 6/72.

  - "Corrosion Studies Using Backscattering Mössbauer Spectroscopy and Electron Energy Resolution." This theoretical study investigates the possibility of obtaining spatially resolved Mössbauer spectra via energy analysis of conversion electrons emitted during the resonance absorption and de-excitation of Fe-57. Application of this technique to the elucidation of oxide-growth processes on austenitic steels are made. Ph.D. Dissertation, started 1971, work near completion.

- Responsibilities at Battelle Memorial Institute:
(5/68-6/71: Supervisor: David Morrison)

Major responsibilities at Battelle, Columbus were with the Chemical Physics Division and involved analysis and modeling studies of processes occurring during loss-of-coolant accidents associated with light water power reactors.

- Reaearch At Euratom:
(4/67-4/68; Supervisor: Rolant Lindner)

Research at Euratom was performed in the Chemistry Department and involved the determination of carbon self-diffusivities in uranium monocarbide. A number of techniques were used, including beta (C-14) ray degradation and specimen sectioning via cathodic sputtering.

- Research at Lawrence Radiation Laboratory, Berkeley:
(6/62-3/67; Supervisor: Donald Olander)

- Literature review and data analysis of reaction mechanisms associated with the gasification of carbon by hydrogen.

- Experimental study of hydrogen dissociation at tantalum surfaces using molecular beam/mass spectrometric techniques (Ph.D. Dissertation).

REVIEWED PUBLICATIONS

1. R. A. Krakowski, and D. R. Olander, "Dissociation of Hydrogen on Tantalum Using Modulated Molecular Beam Technique," J. Chem. Phys. 49, 11, 5027 (1968).

2. R. A. Krakowski, "Self-Diffusion of Carbon in Uranium Monocarbide," J. Nucl. Mater. 32 120 (1969).

3. R. A. Krakowski, D. L. Morrison, and R. L. Ritzman, "An Analytical Description of Oxygen Dissolution into Zircaloy Cladding During a Loss-of-Coolant Accident," Trans. Am. Nucl. Soc. 12 1, 346 (1969).

4. A. Ezis, F. G. Burt, and R. A. Krakowski, "Oxygen Concentration Cell Measurements of Ionic Transport Numbers in PZT Ferroelectrics," J. Am. Cer. Soc. 54, 9, 415 (1971).

5. R. A. Krakowski and R. B. Miller, "An Analysis of Backscatter Mossbauer Spectra Obtained with Internal Conversion Electrons," J. Nucl. Instr. and Meth. 100, 93 (1972).

6. F. L. Ribe, R. A. Krakowski, K. I. Thomassen, and T. A. Coultas, "Engineering Design Study of a Reference Theta-Pinch Reactor (RTPR)," Nucl. Fusion Supplement on Fusion Reactor Design Problems, 99, (1979).

7. T. A. Coultas, R. J. Burke, and R. A. Krakowski, "Environmental and Technological Implications of a Theta-Pinch Fusion Power Plant," 151, ibid.

8. T. A. Coultas, J. E. Draley, V. A. Maroni, and R. A. Krakowski, "Engineering Design Study of a Reference Theta-Pinch Reactor (RTPR): Environmental Impact Study," 169, ibid.

9. J. E. Draley, V. A. Maroni, T. A. Coultas, and R. A. Krakowski, "An Environmental Impact Study of a Reference Theta-Pinch Reactor (RTPR)," Proc. 1st ANS Topical Meeting on the Technology of Controlled Nucl. Fusion, p 564 (April 16-18, 1974).

10. D. J. Dudziak and R. A. Krakowski, "A Comparative Analysis of D-T Fusion Reactor Radioactivity and Afterheat," p 548, ibid.

11. R. A. Krakowski, T. A. Oliphant, and K. I. Thomassen, "Ergonic Optimization and Parametric Study of the RTPR Burn Cycle," p 112, ibid.

12. T. A. Coultas and R. A. Krakowski, "Aspects of Theta-Pinch Power Plant Development," Proc. 5th Symp. on Eng. Prob. of Fusion Reactors (November 6-9, 1973).

13. J. M. Bunch, F. W. Clinard, Jr. D. J. Dudziak, W. V. Green, and R. A. Krakowski, "An Evaluation of Major Material Problems Anticipated for the Reference Theta-Pinch Reactor," ibid.

14. T. A. Coultas, R. A. Krakowski, and P. V. Dauzvardis, "An Engineering Design of a Reference Theta-Pinch Reactor (RTPR)," Trans. Am. Nucl. Soc., 17, 46 (1973).

15. R. A. Krakowski and T. A. Coultas, "Thermal, Mechanical, and Irradiation Response of the Reference Theta-Pinch Reactor (RTPR) First Wall," p 48, ibid.

16. R. A. Krakowski, F. W. Clinard, F. L. Ribe, and T. A. Coultas, "Surface Effects in Controlled Thermonuclear Fusion Devices and Reactors," J. Nucl. Mater. 53, 54 (1974).

17. D. J. Dudziak and R. A. Krakowski, "Radioactivity in a Theta-Pinch Fusion Reactor," J. Nucl. Tech. 25, 32 (1975).

18. R. A. Krakowski, K. I. Thomassen, and T. A. Coultas, "A Technology Assessment of the Reference Theta-Pinch Reactor," Proc. 8th Symp. on Fusion Technology, Noordwijkerhout (The Netherlands), EUR 5182e, p 623 (June 17-21, 1974).

19. K. I. Thomassen, D. J. Dudziak, and R. A. Krakowski, "Prospects for Converting Th to U in a Linear Theta Pinch," Trans. Am. Nucl. Soc. 19, 6 (1974).

20. R. A. Krakowski, R. K. Linford, T. A. Oliphant, R. L. Ribe, and K. I. Thomassen, "Engineering Design of a Fusion Test Reactor (FTR) and a Fusion Engineering Research Facility (FERF) Based upon a Toroidal Theta Pinch," 5th IAEA Conf. on Plasma Physics and Controlled Fusion Research, paper CN-33/G3-3, Tokyo, Japan (1974).

21. R. A. Krakowski, D. J. Dudziak, T. A. Oliphant, and K. I. Thomassen, "An Engineering Design of a Linear Theta-Pinch Hybrid Reactor (LTPHR)," Trans. Am. Nucl. Soc. 21, 61 (1975).

22. R. J. Bartholemew, R. A. Krakowski, and R. L. Hagenson, "Structural Endurance Constraints for High-Field Theta-Pinch Coils," Trans. Am. Nucl. Soc. 22, 20 (1975).

23. W. R. Ellis and R. A. Krakowski, "High-Density Linear Systems for Fusion Power," Proc. 3rd Topical Conf. on Pulsed High-Beta Plasmas, Culham, England (September 9-12, 1975).

24. K. O. Thomassen, R. A. Krakowski, and F. W. Clinard, Jr., "First-Wall Environment in the Reference Theta-Pinch Reactor (RTPR)," J. Nucl. Mater. 63 15 (1976).

25. R. A. Krakowski and F. W. Clinard, Jr., "Thermal-Mechanical Analysis of First-Wall Concepts for the Reference Theta-Pinch Reactor (RTPR)," Trans. Am. Nucl. Soc. 23, 33 (1976).

26. R. L. Hagenson, R. A. Krakowski, and J. G. Martin, "Plasma Engineering and Thermonuclear Burn Dynamics of a Fusion Reactor Based on Reverse-Field Z-Pinch Confinement," Trans. Am. Nucl. Soc. 23 (1976).

27. K. I. Thomassen and R. A. Krakowski, "Burn Cycles for RTPR," Trans. Am. Nucl. Soc. 23, 40 (1976).

28. R. R. Bartsch, R. A. Krakowski, and R. L. Ribe, "The Potential for Fast Feedback Stabilization of the Reference Theta-Pinch Reactor (RTPR)," Proc. IEEE International Conf. on Plasma Science and Fusion Reactor Technology, Austin, Texas (May 24-26, 1976).

29. R. A. Krakowski, R. L. Hagenson, and G. E. Cort, "First-Wall Thermal/Mechanical Analyses of the Reference Theta-Pinch Reactor (RTPR)," Nucl. Technol. 34, 217 (1977).

30. R. A. Krakowski, R. L. Miller, and R. L. Hagenson, "Operating Point Considerations for the Reference Theta-Pinch Reactor (RTPR)," Proc. 2nd ANS Topical Meeting on the Technol. of Controlled Nucl. Fusion, 1, 357 (September 21-23, 1976).

31. L. A. Booth and R. A. Krakowski, "Fusion Electric Generating Station Systems," Proc. IEEE Power Engineering Society Meeting (January 30 - February 4, 1977).

32. R. A. Krakowski, et al., "Pure-Fusion and Fusion-Fission Reactor Applications of High-Density Linear Confinement Systems, Proc. 6th Int. Conf. on Plasma Physics and Controlled Nuclear Fusion, Berchtesgaden FRG (October 6-13, 1976).

33. R. A. Krakowski, W. Quinn, F. L. Ribe, and K. Thomas, "Experiments Towards a Toroidal Theta-Pinch Fusion Reactor," Nucl. Eng. Inter. 45 (February, 1977).

34. F. L. Ribe and R. A. Krakowski, "New Concepts for High-Density Fusion Reactor," Proc. IEEE Conf. on Plasma Science, Rensselaer Polytechnic Institute, Troy, NY (May 23-25, 1977).

35. R. R. Bartsch, R. A. Krakowski, and F. L. Ribe, "Plasma Stabilization Requirments of the Reference Theta-Pinch Reactor (RTPR)," Proc. 9th Symp. on Fusion Technology, 411 (1976).

36. R. L. Hagenson, R. A. Krakowski, and K. L. Thomassen, "A Toroidal Fusion Reactor Based on the Reversed-Field Pinch (RFP)," Proc. IAEA Conf. on Fusion Reactor Design Concepts, 337, paper IAEA-TC-145/20, Madison, WI (October 3-14, 1977) (also Los Alamos Scientific Laboratory report LA-UR-77-2323).

37. R. A. Krakowski, R. W. Moses, R. L. Miller, and R. A. Gerwin, "Fusion Power from Fast Imploding Liners," Proc. IAEA Conf. on Fusion Reactor Design Concepts, 357, paper IAEA-TC-145/21 Madison, Wisconsin (October 3-14, 1977) (also Los Alamos Scientific Laboratory report LA-UR-77-2480).

38. G. E. Cort and R. A. Krakowski, "Heat Transfer in the Lithium-Cooled Blanket of the Reference Theta-Pinch Reactor," Proc. Sixth International Heat Transfer Conference, Toronto, Canada (August, 1978).

39. R. L. Hagenson, R. A. Krakowski, and K. I. Thomassen, "The Reversed-Field Reactor (RFPR)," Trans. Am. Nucl. Soc. 27, 47 (1977).

40. R. L. Hagenson and R. A. Krakowski, "An Engineering Design of a Toroidal Reversed-Field Pinch Reactor (RFPR)," Proc. 10th Symp. on Fusion Technology, Padova, Italy (September 4-8, 1978) (also Los Alamos Scientific Laboratory report LA-UR-78-2322).

41. R. A. Krakowski, R. L. Hagenson, R. L. Miller, and R. W. Moses, "Systems Studies and Conceptual Reactor Designs of Alternative Fusion Concepts at LASL," Proc. 7th International Conf. on Plasma Physics and Controlled Nuclear Fusion Research, IAEA-CN-37/I-2, III, 333-342, Innsbruck, Austria (1978) (also Los Alamos Scientific Laboratory report LA-UR-78-1962).

42. A. S. Tai and R. A. Krakowski, "Energy-Balance and Economic Considerations for Fusion Thermochemical Hydrogen Production," Trans. Amer. Nucl. Soc. 30, 42 (1978).

43. R. L. Hagenson and R. A. Krakowski, "A Cost-Constrained Design Point for the Reversed-Field Pinch (RFPR)," Proc. 3rd ANS Topical Meeting on the Technology of Controlled Nuclear Fusion, Santa Fe, New Mexico (May 9-11, 1978).

44. G. E. Gryczkowski, R. A. Krakowski, L. C. Steinhauer, and J. Zumdieck, "First-Wall Design Limitations for Linear Magnetic Fusion (LMF) Reactors," ibid (also Los Alamos Scientific Laboratory report LA-UR-78-1970).

45. L. A. Booth, M. G. Bowman, G. E. Cort, K. E. Cox, D. J. Dudziak, R. A. Krakowski, J. H. Pendergrass, and A. S. Tai, "Production of Electro-Thermochemical Hydrogen Using a Fusion Source of High-Temperature Process Heat," ibid (also Los Alamos Scientific Laboratory report LA-UR-78-1459).

46. R. W. Moses, R. A. Krakowski, and R. L. Miller, "Fast-Imploding Linear Fusion Power," ibid (also Los Alamos Scientific Laboratory report LA-UR-78-1369).

47. R. A. Krakowski, "A Survey of Linear Magnetic Fusion Reactors," ibid (also Los Alamos Scientific Laboratory report LA-UR-78-1319).

48. R. L. Miller and R. A. Krakowski, "Thermal Conduction and Alpha-Particle Constraints for the Ignition of a D-T Linear Magnetic Fusion (LMF) Reactor," Nucl. Fusion, 18, 12 (1978).

49. L. A. Booth and R. A. Krakowski, "Fusion Electric Generating Station Systems," Proc. Winter Meeting of IEEE Power Engineering Society (January 30 - February 4, 1977).

50. R. A. Krakowski, K. E. Cox, J. H. Pendergrass, and L. A. Booth, "Synfuel (Hydrogen) Production from Fusion Power," 14th Intersociety Energy Conversion Engineering Conference, 2, 1544, Boston, MA, (August 5-10, 1979).

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53. R. L. Hagenson and R. A. Krakowski, "The Reversed-Field Pinch Reactor," Trans. Amer. Nucl. Soc. 33, 96 (1979) (also Los Alamos Scientific Laboratory report LA-UR-79-1640).

54. R. A. Krakowski and R. L. Hagenson, "Plasma Simulations of the ELMO Bumpy Torus Reactor (EBTR)," Trans. Amer. Nucl. Soc. 33, 38 (1979) (also Los Alamos Scientific Laboratory report LA-UR-79-1641).

55. R. L. Miller, R. A. Krakowski, and C. G. Bathke, "TORMAC Fusion Reactor Design Points," Trans. Amer. Nucl. Soc. 33, 98 (1979) (also Los Alamos Scientific Laboratory report LA-UR-79-1679).

56. C. G. Bathke, R. A. Krakowski, and H. S. Cullingford, "Tritium Transport in a Packed-Bed Li_2O Fusion Blanket," Trans. Amer. Nucl. Soc. 33, 72 (1979) (also Los Alamos Scientific Laboratory LA-UR-79-1707).

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REPORTS

1. R. A. Krakowski, "Dissociation of Hydrogen or Tantalum Using Modulated Molecular Beam Technique," USAEC Rept. UCRL-17736 (1967).

2. R. A. Krakowski, "Sputtering Characteristics of Uranium Monocarbide and Application to Diffusion Studies," Rapport du Stage, EURATOM (Ispra) Rept. #1384, Brussels, Belgium (1969).

3. R. A. Krakowski, and D. R. Olander, "Survey of the Literature on the Carbon-Hydrogen System," USAEC Rept. UCRL-19149 (1970).

4. R. A. Krakowski, F. L. Ribe, and T. A. Coultas, "An Engineering Design Study of a Reference Theta-Pinch Reactor (RTPR)," USAEC Rept. LA-5336/ANL-8019 (1974).

5. T. A. Coultas, J. E. Draley, J. E. Maroni, and R. A. Krakowski, "An Engineering Design of a Reference Theta-Pinch Reactor (RTPR): An Environmental Impact Study," USAEC Rept. LA-5336/ANL-8019, Vol. II (1975).

6. R. A. Krakowski, W. V. Green, and W. C. Turner, "LASL's Experience in Welding Molybdenum," USAEC Rept. LA-UR-73-505 (1973).

7. R. A. Krakowski, and D. J. Dudziak, "Prospects for Converting ^{232}Th to ^{233}U in a Linear theta-Pinch Hybrid Reactor (LTPHR)," USERDA Rept. ERDA-4 (1975).

8. R. A. Krakowski, "Blanket and Shielding Technology Assessment of the Reference Theta-Pinch Reactor (RTPR)," Proc. Magnetic Fusion Energy Blanket and Shielding Workshop, March 29-April 12, 1976, USERDA Rept. ERDA-76/117/1. USERDA Rept. LA-UR-76-646 (1976).

9. P. E. Armstrong, and R. A. Krakowski, "Thermal Shock Experiment (TSEX)-A "Proof-of-Principle" Evaluation of the Use of Electron-Beam Heating to Simulate Anticipated for the First-Wall of the Reference Theta-Pinch Reactor (RTPR)," USERDA report LA-6861-MS (June, 1977).

10. R. A. Krakowski, R. L. Miller, and R. L. Hagenson, "Engineering and Physics Considerations for a Linear Theta-Pinch Hybrid Reactor," US/USSR Symp. on Fusion-Fission Reactors, 155, July 13-16, 1976 (CONF-760733).

11. A. S. Tai and R. A. Krakowski, "A Simple Economic Parametrics Analysis of Fissile-Fuel Production by Fusion-Fission Reactors," Proc. 2nd MFE Fusion-Fission Energy Systems Review Meeting, Washington, DC (November 2-3, 1977) (also Los Alamos Scientific Laboratory report LA-UR-77-2879).

12. R. A. Krakowski, "First Wall Surface Problems Anticipated for Fusion Reactors," ACS Summer Symposium, Phoenix, Arizona (1976).

13. J. P. Freidberg, R. A. Krakowski, W. E. Quinn, and R. E. Siemon, "Review of the Linear Theta-Pinch (LTP) Concept," USERDA Rept. LA-UR-77-0961 (1977).

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15. R. L. Miller and R. A. Krakowski, "Fusion Reactor Plant Design for the Linear Theta-Pinch (LTPR)," Los Alamos Scientific Laboratory report LA-UR-78-2296 (1978).

16. R. L. Hagenson and R. A. Krakowski, "Fusion Reactor Design for the Reversed-Field Pinch Reactor (RFPR)," Los Alamos Scientific Laboratory report LA-UR-78-2268 (1978).

17. A. S. Tai and R. A. Krakowski, "Generalized Energy-Balance and Economic Considerations for Thermochemical Hydrogen Production from Fusion Reactor Blankets," Los Alamos Scientific Laboratory report LA-UR-78-1833 (1978).

18. R. A. Krakowski and R. W. Moses, "Energy-Balance and Blast Containment Considerations for FLICR," Los Alamos Scientific Laboratory report LA-UR-77-1276.

19. A. S. Tai and R. A. Krakowski, "Non-Electric Uses of Thermal Energy Generated in the Production of Fissile Fuel in Fusion-Fission Reactors: A Simple Economic Parametric Analysis," Proc. US/USSR Symp. on Fusion-Fission (Hybrid) Reactors, Princton, New Jersey (January 22-26, 1979) (also Los Alamos Scientific Laboratory report LA-UR-79-84).

20. R. L. Miller and R. A. Krakowski, "Magnetic Fusion Alternate Concepts Heating Requirements," Los Alamos Scientific Laboratory report LA-UR-78-15 (1978).

21. J. M. Williams, L. A. Booth, and R. W. Krakowski, "Overview of Systems Requirements for Impact Fusion Power," Proc. of the Impact Fusion Workshop, 44-64, Los Alamos Scientific Laboratory report LA-8000-C (August, 1979).

22. R. A. Krakowski, R. W. Moses, and J. D. Jacobson, "Blast Confinement Computations for the Fast-Liner Reactor (FLR)," ibid, 107-127 (also Los Alamos Scientific Laboratory report LA-UR-79-2035).

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24. R. W. Moses, R. A. Krakowski, and R. L. Miller, "A Conceptual Design of the Fast-Liner Reactor (FLR) for Fusion Power," Los Alamos Scientific Laboratory report LA-7686-MS (February, 1979).

25. R. L. Miller, R. A. Krakowski, and C. G. Bathke, "A Parametric Study of the Tormac Fusion Reactor Concept," Los Alamos Scientific Laboratory report LA-7935-MS (August, 1979).

26. R. L. Hagenson, R. A. Krakowski, and G. E. Cort, "The Reversed-Field Pinch Reactor (RFPR) Concept," Los Alamos Scientific Laboratory report LA-7973-MS (August 1979).

27. R. A. Nebel, R. L. Hagenson, R. W. Moses, and R. A. Krakowski, "Comparison of Zero-Dimensional and One-Dimensional Thermonuclear Burn Computations for the Reversed-Field Pinch Reactor (RFPR)," Proc. of the Varenna School and Workshop on Plasma Physics Close to Thermonuclear Conditions, Varenna, Italy (August 26-September 7, 1979) (also Los Alamos Scientific Laboratory report LA-8185-MS January 1980).

28. R. A. Krakowski and R. L. Hagenson, "Plasma Simulation of the EBT using Kovrizhnykh Neoclassical Transport," Proc. of the EBT Transport Workshop, 1979, Washington, DC, DOE report (also Los Alamos Scientific Laboratory report LA-UR-79-1438).

29. R. L. Hagenson, R. A. Krakowski, A. S. Tai, and R. W. Moses, "The Dense Z-Pinch (DZP) as a Fusion Power Reactor: Preliminary Scaling Calculations and Systems Energy Balance," Los Alamos Scientific Laboratory report LA-8186-MS (January 1980).

30. A. S. Tai and R. A. Krakowski, "Non-Electric Uses of Thermal Energy Generated in the Production of Fissile Fuel in Fusion-Fission Reactors: A Comparative Economic Parametric Analysis for a Hybrid With or Without Synthetic Fuel Production," Proc. US/USSR Symp. on Fusion-Fission (Hybrid) Reactors, Princeton, NJ (January 22-26, 1979) (also Los Alamos Scientific Laboratory report LA-UR-79-574).

31. A. Andrade, R. A. Krakowski, and C. G. Bathke, "Steady-State Microwave Power Requirements for EBTR Electron Rings," Proc. EBT Ring Workshop, Oak Ridge, TN (December 3-5, 1979).

32. R. L. Hagenson and R. A. Krakowski, "Preliminary Reactor Scaling for a Compact Torus Fusion Reactor (CTOR)," Los Alamos Scientific Laboratory report LA-8448-MS (August, 1980).

33. R. L. Miller and R. A. Krakowski, "Re-assessment of the Slowly-Imploding Liner (Linus) Fusion Reactor Concept," Los Alamos Scientific Laboratory report (to be published, 1980).

34. G. E. Gryczkowski and R. A. Krakowski, "Plasma/Wall Interactions in Dense Linear Magnetic Fusion (LMF) Reactors: Evaporating First Walls," Los Alamos Scientific Laboratory report (to be published).

35. R. A. Krakowski and C. G. Bathke, "Re-assessment of the Elmo Bumpy Torus (EBTR) Fusion Reactor Concept: Phase I, Physics Design," Los Alamos Scientific Laboratory report (to be published, 1980).

36. R. L. Hagenson and R. A. Krakowski, "Time-Dependent Plasma Simulations and Energy Balance for a Compact Torus Fusion Reactor (CTOR)," (to be published, 1980).

37. R. L. Hagenson, and R. A. Krakowski, "Physics Considerations of the Reversed-Field Pinch Fusion Reactor," Proc. RFP Workshop, Los Alamos, New Mexico (April 28 - May 2, 1980) (also Los Alamos Scientific Laboratory report LA-UR-80-2219).

G. EDWARD CORT

EDUCATION

B.S. Mechanical Engineering, Carnegie-Mellon University, 1960
M.S. Nuclear Engineering, Carnegie-Mellon University, 1960.
Short Courses:
- Advanced FORTRAN
- Analyzing and Interpreting Industrial Experiments
- Mechanical Design Reliability
- Computer Codes for Heat and Fluid Flow
- Design and Control of Technical Presentations
- Numerical Modeling of Detonations

Recent Postgraduate Courses for Credit:
- Higher Mathematic for Engineers
- Seminar in Advanced Fluid Mechanics
- Topics in Finite Element Methods
- Physical Chemistry

Registered Professional Engineer, Pennsylvania.

SUMMARY

Specialist in compressible and incompressible fluid mechanics, heat transfer, finite element methods, and development of numerical methods for heat transfer. Experienced in the engineering analysis of heat, mass, and momentum transfer for a wide variety of LASL projects. Currently Section Leader for six staff members engaged in fluid flow and heat transfer modeling and analysis.

EXPERIENCE

<u>May 1974 to Present</u>

Staff Member, Los Alamos Scientific Laboratory. Analyze heat transfer and predict fission product release from High Temperature Gas-Cooled Reactor (HTGR) in Loss of Forced Circulation Accident. Assessment of design change for a Very High Temperature Reactor (VHTR) for process heat applications, parametric one-dimensional supersonic flow and heat transfer in deuterium flow loop for 14 MeV Intense Neutron Source, transient temperature response in fuel elements in liquid-metal-cooled fast reactor safety test facility, thermal stress analysis of vacuum wall for controlled fusion Reference Theta Pinch Reactor (TRPR), blanket design for RTPR and for production of thermochemical hydrogen, heat and mass transfer in underground coal gasification, evaluation of effects of sabotage on nonpower reactors.

<u>August 1972 to May 1974</u>

On assignment to Los Alamos Scientific Laboratory as Industrial Staff Member for Nuclear Rocket and Subterrene Projects. Carried out thermal-hydraulic analysis of Rover reactor axial support system and core periphery, using coupled transient analysis codes for finite difference heat conduction and convection in multiple parallel flow channels. Analyzed performance and predicted effects of design modifications in Subterrene rock melting penetrators. Assessed experimental results with regard to penetration rate, melting efficiency, failure modes, glass thermal treatment, etc., and recommended design improvements.

February 1963 to August 1972

Engineer and Senior Engineer, Westinghouse Electric Corporation, Astronuclear Laboratory, Pittsburgh, Pennsylvania. Lead Engineer responsible for thermal and hydraulic design of NERVA reactor fuel elements: develop methods for probabilistic analysis of maximum fuel temperatures and corrosion; review fuel design and recommend changes; plan, conduct, and coordinate parametric trade studies to optimize the fuel design; present results at customer meetings and design reviews. Member of Westinghouse Test Monitor Team for MRX-A6 reactor tests in Nevada. Responsible for thermal and hydraulic conceptual design studies of a helium-cooled graphite-moderated reactor for use in a Brayton cycle power system. Independently carried out thermal and hydraulic design studies for a special purpose nitrogen-cooled reactor. On temporary assignment to Westinghouse Nuclear Energy Systems, reviewed and critiqued heat transfer and fluid flow modeling assumptions used in two PWR safety analysis computer codes, including comprehensive study of experimental data. Responsible for selecting fuel elements and specifying flow orificing to achieve optimum temperature flattening. With technician assistant, responsible for providing assembly instructions for fuel clusters for five NERVA reactors.

AWARDS AND SOCIETIES

AEC Fellowship, Pi Tau Sigma Mechanical Engineering Honorary, Sigma Xi.

PUBLICATIONS

1. "Application of Coupled Fluid Flow and Heat Conduction Analysis to NERVA Reactors," presented at AIAA Joint Propulsion Specialists Conference, June 1962. (Co-author)

2. "Third Generation Computer Used to Control Fuel Element Assembly," presented at ANS Conference on the Effective Use of Computers in the Nuclear Industry, April 1969. (Principal Author)

3. "Rock Heat-Loss Shape Factors for Subterrenen Penetrators," Los Alamos Scientific Laboratory report LA-5435-MS (1973).

4. "Deuterium Flow Loop for a Supersonic Gas-Target Intense Neutron Source," Los Alamos Scientific Laboratory report LA-5849-MS (1975).

5. "Conceptual Design for a Fast Reactor Safety Test Facility," Los Alamos Scientific Laboratory report LA-6031-MS (Co-author).

6. "Predicted Nuclear Heating and Temperatures in Gas-Cooled Nuclear Reactors for Process Heat Applications," Los Alamos Scientific Laboratory report LA-6113-MS (Principal author).

7. "Development of Coring, Solidating, Subterrene Penetrators," Los Alamos Scientific Laboratory report LA-6265-MS (Co-author).

8. "Core Heatup and Fission Product Release from an HTGR Core in an LOFC Accident," Los Alamos Scientific Laboratory report LA-NUREG-6499-MS (Principal author).

9. "First Wall Thermal-Mechanical Analyses of the Reference Theta-Pinch Reactor," Nuclear Technology, V. 34, July 1977 (Co-author)

10. "Development of Pyrolytic Graphite/Silicon Carbide Composite Materials for Rocket-Nozzle Applications," V. I, II, and III, Los Alamos Scientific Laboratory reports LA-UR-77-2042, -2444, and -2679. To be published by the Air Force Rocket Propulsion Laboratory as a technical report (TR). (Co-author).

11. "Heat Transfer through Coals and Other Naturally Occurring Carbonaceous Rocks," Proceedings of 15th International Thermal Conductivity Conference, Ottawa, Canada, August 24-26, 1977. (Co-author).

12. "Heat and Mass Transfer through Southwestern Subbituminous Coals," Proceedings of Third Underground Coal Extraction Symposium, Stanford Sierra Lodge, June 6-10, 1977. (Co-author).

13. "Heat Transfer in the Lithium-Cooled Blanket of a Pulsed Fusion Reactor," to be published in Proceedings of 6th International Heat Transfer Conference, Toronto, Ontario, Canada, August 7-11, 1978. (Principal author)

Thomas A. Butler, P.E.

Staff Member
Group Q-13
Reactor Technology and Advanced Heat Transfer

Education: B.S. - Colorado State University, M.S.A.E. - University of Michigan

Experience: 4 years at LASL
10 years total

LANL Positions - Mr. Butler has been involved in the thermomechanical and structural analysis of reactor components exposed to both normal operating and accident conditions. Some of these components are very complex geometrically and are typically exposed to combinations of surface connection, conduction, and radiation heat loads as well as presure and gravity loads. The response is generally nonlinear. He was principle investigator on a project to develop analytical methods for evaluating nuclear material shipping containers when they are subjected to accident conditions. This involved the calculation of highly nonlinear material and geometrical response. Mr. Butler participated in several experimental programs with other national laboratories to obtain substantiating data for his modeling work.

Previous Associations - At the Lockheed Missles and Space Company Mr. Butler was involved in the analysis of the structural response of spacecraft to launch loads and on-orbit excitations. He was also involved in developing analytical techniques for performing these calculations. Calculations were normally correlated with either actual launch and orbital data or ground tests specifically designed to verify analytical models. Mr. Butler was also directly involved in liaison with subcontractors, particularly in the area of equipment qualification for vibration environments.

PUBLICATIONS

T. A. Butler and C. A. Anderson, "Three-Dimensional Transient Thermoelastic Analysis of a Graphite Core Support Block," Los Alamos Scientific Laboratory report (to be published, 1981).

T. A. Butler and J. G. Bennett, "Nonlinear Response of a Post-Tensioned Concrete Structure to Static and Dynamic Internal Pressure Loads," Computer and Structures for ADINA Conference (June 10-12, 1981).

R. J. Bartholomew and T. A. Butler, "Analysis of Railcar - Shipping Container System Response to Impact Conditions," Los Alamos Scientific Laboratory report LA-8122-MS (January 1980).

T. A. Butler, "The Effects of Drop Testing of Scale Model Shipping Containers Shielded with Depleted Uranium," Los Alamos Scientific Laboratory report LA-8120-MS (February 1980).

T. A. Butler, E. G. Endebrock, and J. B. Payne, "CRASHC - A Two-Dimensional Code to Compute the Response of Axisymmetric Shipping Containers to End-On Impacts," Los Alamos Scientific Laboratory report LA-8121-MS (January 1980).

T. A. Butler, "Effect of Attitude Control Thruster Induced Structural Vibrations on Sensed Accelerations of a Spare Vehicle," Los Alamos Scientific Laboratory report LA-6965-MS (September 1977).

T. A. Butler, "Model Synthesis Computer Program MOSTNZ," Lockheed Missles and Space Co. report LMSC/D384414 (December 1973).

DOVE, RICHARD C.

I. Education

BSME--IOWA STATE UNIVERSITY, 1947
MSME--Iowa State University, 1949
Ph.D. (T. and A.M.)--Iowa State University, 1954

Honor Graduate in Mechanical Engineering, 1947
Westinghouse Fellow in Mechanical Engineering, 1948-49
National Science Fellow, 1952-54

II. Work Record

Teaching:
- Instructor, UNM - 1947-48, 1949-50
- Assistant Professor, UNM - 1951-52
- Assistant Professor, ISU - 1954-55
- Associate Professor, UNM - 1955-60
- Professor, UNM - 1960-1975
- Lecturer, U. of California (Berkeley) - Fall, 1965
- Visiting Professor, National Uni. of Engineering, Lima, Peru - June-July, 1971

Other University Service:
- Chairman, M.E. Department - 1963-68
- Dean, Engineering - 1968-74
- Have served on University Policy Committee
- Research Committee; Greater UNM Fund Committee, Chairman
- Committee on the University; and Extension Committee

Industrial:
- Sandia Corporation - 1952, '55, '56, '61, '62, '64, '65, (Summer)
- Los Alamos Scientific Laboratory, Full time, 1974 - present
- Have served as Consultant to:
  - Sandia Corporation - 1951-52, 1964 to 1968
  - ACF Industries - 1959-60
  - Los Alamos Scientific Laboratory - 1956-1968
  - USAF Base, Holloman - 1964

Research:
- Director of the following projects:
  - "Effect of Rate of Loading on Mechanical Properties" for LASL - 1957-59
  - "Strain Distribution in a Thin Circular Disk," for Sandia Corp. k- 1960-61
  - "Measurement of Strain at Interior Points," for Sandia Corp. - 1959-61
  - "Dynamic Stress-Strain Diagrams of Filled Epoxy Plastics," for Sandia Corp. - 1960-61
  - "Transient Thermal Stresses in Cylindrical Regions" (co-director with Dr. Ju) for Sandia Corp. - 1959-61
  - "Instrumentation for Rocket Sleds," for USAF, Holloman AFB - 1964-65
  - "Investigation of Contact Fusing," for AFSWL, Kirtland AFB - 1966-68

III. Publications

"Determination of the Effective Strained Length of Standard Stud Bolts," ASME Pre-print paper given at spring meeting ASME, Atlanta, Georgia, April 1951 (co-author).

"Strain Measurement Errors in Materials of Low Modulus," Vol. 81, Separate #691, Proc. ASCE May 1955.

"Experimental Technique for Predicting the Dynamic Behavior of Rubber," Vol. 77, No. 6 Trans. ASME, August 1955 (co-author).

"The Use of Electrical Resistance Strain Elements in Three Dimensional Stress Analysis," Experimental Mechanics, Vol. 1, No. 6, June 1961 (co-author).

"A Practical Approach to Shock Mounting," 48th Shock and Vibration Bulletin, POart III - 1960 (co-author).

"Obtaining Stress-% Compression Diagram of Foamed Plastics at High Rates of Compression," ASME paper, Number 6D-RP-9 (co-author).

"Data Presentation for Cushioning Materials," presented at Soc. of Plastics Engineers annual meeting, January 1961, Washington, DC (co-author).

"Similitude in Package Cushionings," ASME Paper No. 61-WA-50, published in Journal of Applied Mechanics (co-author).

"Selection of Gages for Strain Measurement at Interior Points," Experimental Mechanics, Vol. 2, No. 6, June 1962 (co-author).

"Measurement of Interior Strains in a Bar Subjected to Longitudinal Impact," Experimental Mechanics, Vol. 2, No. 10, October 1962 (co-author).

"Transient Calibration of Piezoelectric Accelerometers," Journal of Environmental Sciences, October 1962 (co-author).

"Strain Distribution in a Short Force Link," Journal of Environmental Sciences, April 1963 (co-author).

"Construction and Evaluation of a Three Dimensional Strain Rosettes," Experimental Mechanics, Vol. 3, No. 9, September 1963 (co-author).

"Instrumentation for Shock Motion Measurement," Proceedings of the Institute of Environmental Sciences, 1963.

"Electric Resistance Strain Gage Characteristics and Types," SCTM 363-61(73), March 1962 (co-author).

"Reduction of Strain Gage Data," SCTm 362-61 (73), January 1962 (co-author).

"Circuitry and Recording Equipment used with Variable-Resistance Metallic Strain Gages," SCTM 364-61(73), September 1962 (co-author).

"Electrical Resistance Strain Gages Applied to Special Problems," SCTM 365-61(73), August 1962 (co-author).

"Experimental Stress Analysis and Motion Measurement," Dove and Adams, 505 pages, Charles E. Merrill 1964 (Graduate level text).

"Calibration and Evaluation of Accelerometers in the 10,000g to 100,000g Range," ISA Pre-print 17.3-1-65 (co-author).

"High Frequency Problems in the Transmission of Piezoelectric Transducer Data," ISA Journal, Vol. 13, No. 1, January 1966 (co-author)

"The Selection and Evaluation of Shock Test Instrumentation," Institute of Environmental Sciences Tutorial Lecture Series - Dynamics, 1966, pp. 115-141 (co-author)

"Circuitry for Conditioning the Transducer Output Signal," ISA Journal, Vol. 14, No. 4, April 1967. This paper won a $500.00 award as the best technical paper published by ISA during 1967.

"Wave Propagation in a Thin Hollow Cone by a Finite Element Method," Journal of Sound and Vibration, Vol. 24, No. k2, pp. 211-218, 1972 Valathur, Dove, Albrecht

"Numerical Correction of Transient Measurements," ISA Trans. Vol. 12, No. 3, pp. 286-295, 1973 - Bickle and Dove

"Proposal for Analysis of HTGR Core Response to Seismic Input," Bennett and Dove, LA-5821-MS, Los Alamos Scientific Laboratory, January 1975

"Seismic Response of a Block Type Nuclear Reactor Core," Dove, Bennett, Merson, LA-NUREG-6377-MS, Los Alamos Scientific Laboratory, January 1975.

"A Proposal for a Seismic Facility for Reactor Safety Research," Anderson, Dove, Rhorer, LA-NUREG-6388-P, Los Alamos Scientific Laboratory, 1976

IV. Professional Societies

Registered Professional Engineer (New Mexico) 1951-present
Pi Tau Sigma
Sigma Tau
Tau Beta Pi
Phi Kappa Phi

V. Personal

Birth - November 29, 1924; Fairbury, Nebraska
Married - Three Children - all over 21
Served in U.S. Navy 1943-1945, Honorable Discharge

VI. Honors and Awards

Western Electric award for Outstanding Engineering Educator, April, 1966, presented by the Southwest Section of the American Society for Engineering Education

Award for best technical paper published by the Instrument Society of America during 1967. (See item II - Publications)

Annual Research Lecturer, The University of New Mexico, April 1966.

APPENDIX N

APPENDIX N

EG&G/GENERAL ATOMIC COMPANY RESPONSE

Following are Appendix H from the GA-EG&G Idaho expression of interest for TPE-II and an extract (pp 36-41) from our proposal on that program which describes the tungsten mesh furnace facility. Photographs have been omitted.

H. FACILITIES AND RESOURCES

H.1 BENCH SCALE FACILITIES

The capabilities available at General Atomic and EG&G for bench scale thermal-hydraulic, mechanical and materials science and engineering testing include a broad range of skills and facilities suitable for evaluation and testing of metallic and ceramic materials for a variety of sophisticated applications. These capabilities are applicable to the evaluation of equipment used in a wide variety of temperature environments found in power generation facilities, petrochemical plants, advanced energy conversion systems, etc. and will be directly applicable to TPE II.

A full complement of laboratory facilities and equipment developed for GA's HTGR, GCFR and Fusion programs, and for EG&G's LWR and LMFBR program and the availability of many scientific and engineering personnel (such as Physicists, Chemists, Mechanical Engineers, Stress Analysts, etc.) enables the testing and materials groups effectively to perform programs that require interdisciplinary skills.

Some examples capabilities available are as follows:

Creep, rupture, fatigue, creep-fatigue and related behaviors can be studied in elevated temperature, mechanical properties and behavior test facilities. Some current programs emphasize the effects of unusual environments (such as helium and liquid metals) on these properties of both metals and ceramics. Since much of this work is carried out to support the design of nuclear components for sophisticated Codes (such as ASME Code Case N-47), the data generated can be given the most exacting analysis.

Corrosion behavior in elevated temperature gaseous environments capabilities includes the ability not only to perform testing but also to analyze resulting data in terms of kinetic and thermodynamic models.

Corrosion behavior in moderate temperature steam and water environments capabilities include evaluation of general corrosion, erosion corrosion, stress corrosion cracking, etc.

Friction wear and fretting behavior evaluations can be performed in a variety of test facility environments. These test capabilities are supplemented by facilities for producing specialized coatings for physical vapor deposition, chemical vapor deposition and ion plating methods and by analytical staff capable of performing required associated flow-induced vibration and related studies.

Fabrication and joining studies can be performed in laboratories and similar facilities using both conventional and unconventional welding and brazing equipment and techniques. The effects of fabrication-induced cold work on materials behavior can be evaluated along with the effects of residual stresses. Studies of the spatial distribution of properties in both metallic and ceramic materials can be performed.

Microstructural characteristics of materials can be evaluated in great depth by use of the electron microscope, scanning microscope, microprobe analyzer, X-ray diffraction and related analytical equipment available at GA and EG&G. Expert staff provide complete interpretations and models of microstructure development in both metallic and ceramic materials.

H.2 WATER TEST FACILITIES

EG&G operates several high pressure and temperature water test loops in support of its fission reactor test programs. Major facilities at EG&G of this kind are listed in Table H.2-1. One or more of these facilities may be used in connection with TPE II. Several of these facilities will be described in additional detail to illustrate the range of capability.

TABLE H.2-1
E.G.&G. TEST LOOPS

Loop	Location	Type	Pressure MPa	Temp. ^{o}C	Flow Rate m^3/s	Status
Ambient Flow Calibration	ARA-3	Water	2.1	52	0.017	Operational
MTR/ETR Test Loop	TRA	Water	1.8	65	0.038	Operational
Ballistic Flow Calibration	ARA-3	Water	0.8	65	0.076	Operational
Shim Rod Loop	TRA	Water	3.2	115	0.006	Operational
ATR Safety Rod Drive	TRA	Water	3.2	115	0.05	Operational
High Temperature Hydraulic Test	TRA	Water	7.7	260	0.076	Operational
Air/Water	TAN	Air/Water	2.2	93	1.41/.05	Operational
High Temperature	ARA-3	Water/Steam	15.6	343	0.019	Operational
Fast Loop	TAN-LTSF	Water	7.8	279	0.303	Operational
Blowdown Facility	TAN-LTSF	Water	15.6	288	0.32 m^3 Blowdown	Operational
Two-Phase Flow Loop	TAN-LTSF	Water/Steam	6.0	275	0.445	Operational

Two-Phase Flow Loop

The EG&G Two Phase Flow Loop (TPFL) is a large, high temperature steam-water test facility. This loop, shown schematically in Fig. H.2-1, consists of four large steam supply vessels that produce steam by controlled flashing, a moisture separator, an independent pumped and metered water loop, a two-phase mixing section and a steam separator that permits operation with a range of steam-water mixtures. Loop operating characteristics are given in Table H.2-2.

TPFL instrumentation includes loop instrumentation, as well as a tomographic densitometer, traversing impedance probes, pitot tubes, and six beta/gamma densitometers.

Two-phase flow conditions can be maintained at constant pressure and temperature until the pressure in the vessels can no longer drive the steam through the system (approximately 4 minutes at maximum steam flow conditions at 6.7 MPa (1000 psi) and proportionally longer at lower flows). Tests can also be conducted with subcooled or saturated, single phase, high temperature water or single phase saturated steam passing through the test section.

Blowdown Loop

The blowdown loop shown on Fig. H.2-2 is an experimental apparatus designed for instrument calibration, component testing and heat transfer experiments in simulated PWR steady-state and transient environments. It operates up to a steady state pressure of 15.5 MPa (2250 psi) and up to a temperature of 556 K (550 F), and has a volume of 0.322 m^3 (85 gal). The system includes three intercoupled loops built around a 0.277 m^3 (60 gal) pressure vessel and has a quick-opening blowdown valve. In this system, two different basic configurations, a small test loop and a large test loop, can be blown down. The system consists of all flanged piping, and major modifications are routinely made in only a few days. During a two-shift day of testing, three to five short duration blowdowns and one complete system blowdown can be completed.

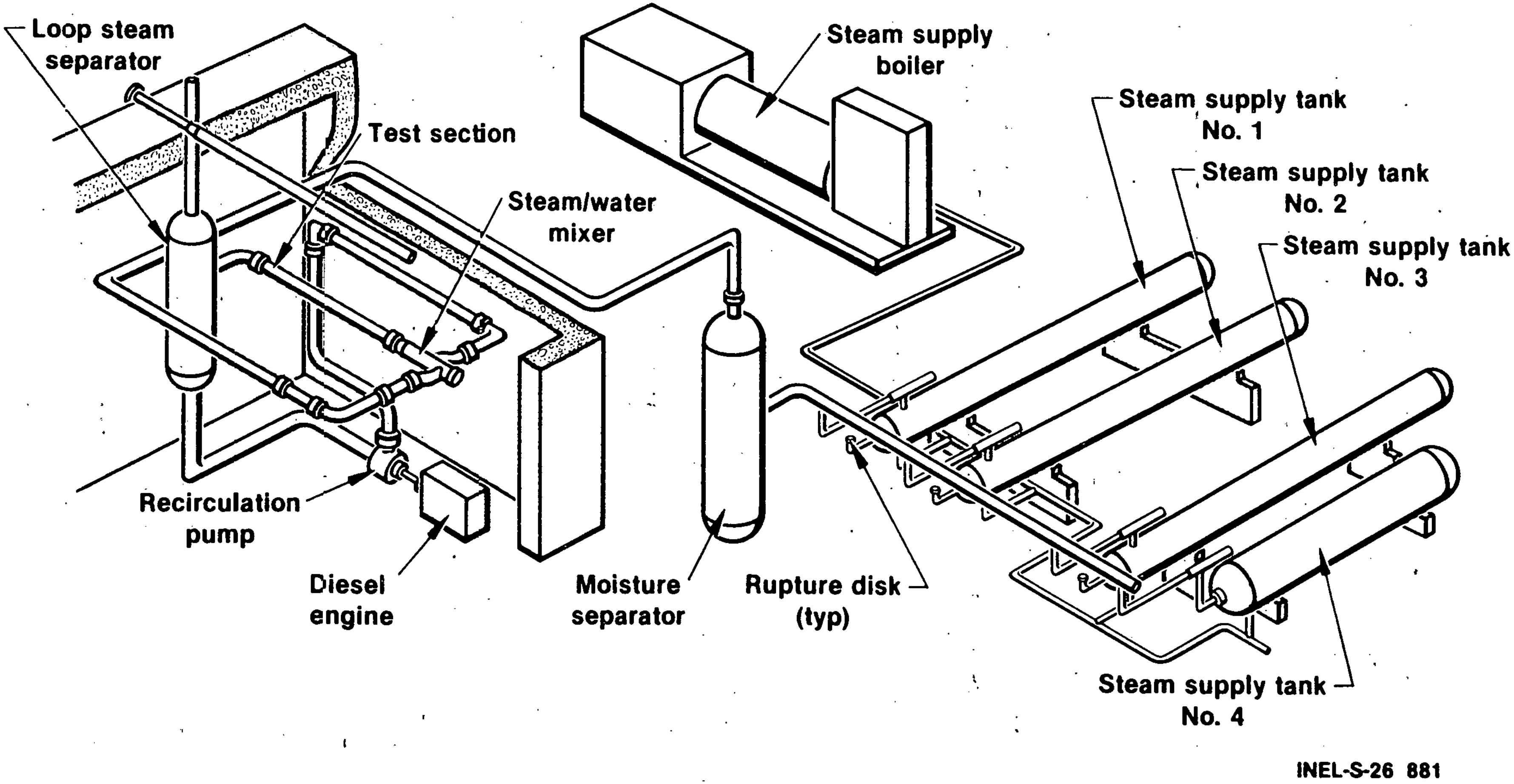


Fig. H.2-1 Two Phase Flow Loop

TABLE H.2-2
EG&G TWO PHASE FLOW LOOP DESIGN FEATURES

	SI	English
1. GENERAL CHARACTERISTICS		
Maximum pressure	6.9 MPa	1000 Psi
Maximum steam mass flow rate	25 kg/s	55 lbm/s
Maximum circulating water mass flow rate	420 kg/s	924 lbm/s
Maximum pressure drop for the steam	3 MPa	435 Psi
Total volume of loop	110 m^3	3885 ft^3
Water volume	15 m^3	530 ft^3
Steam supply vessels volume	85 m^3	3000 ft^3
2. LIQUID SUPPLY SYSTEM		
Maximum volume flow rate	2100 m^3/h	9000 GPM
Pump head at 9000 GPM	73 m	940 ft
NPSH required at 9000	4 m	13.1 ft
Power of diesel drive motor	522 kW	700 Hp
3. STEAM SUPPLY SYSTEM		
Steam Generator Maximum thermal power	3.7 MW	3500 Btu/s
Storage tank steam supply	85 m^3 of Sat Water 10.4 MPa	3000 ft of Sat Water 1500 Psi

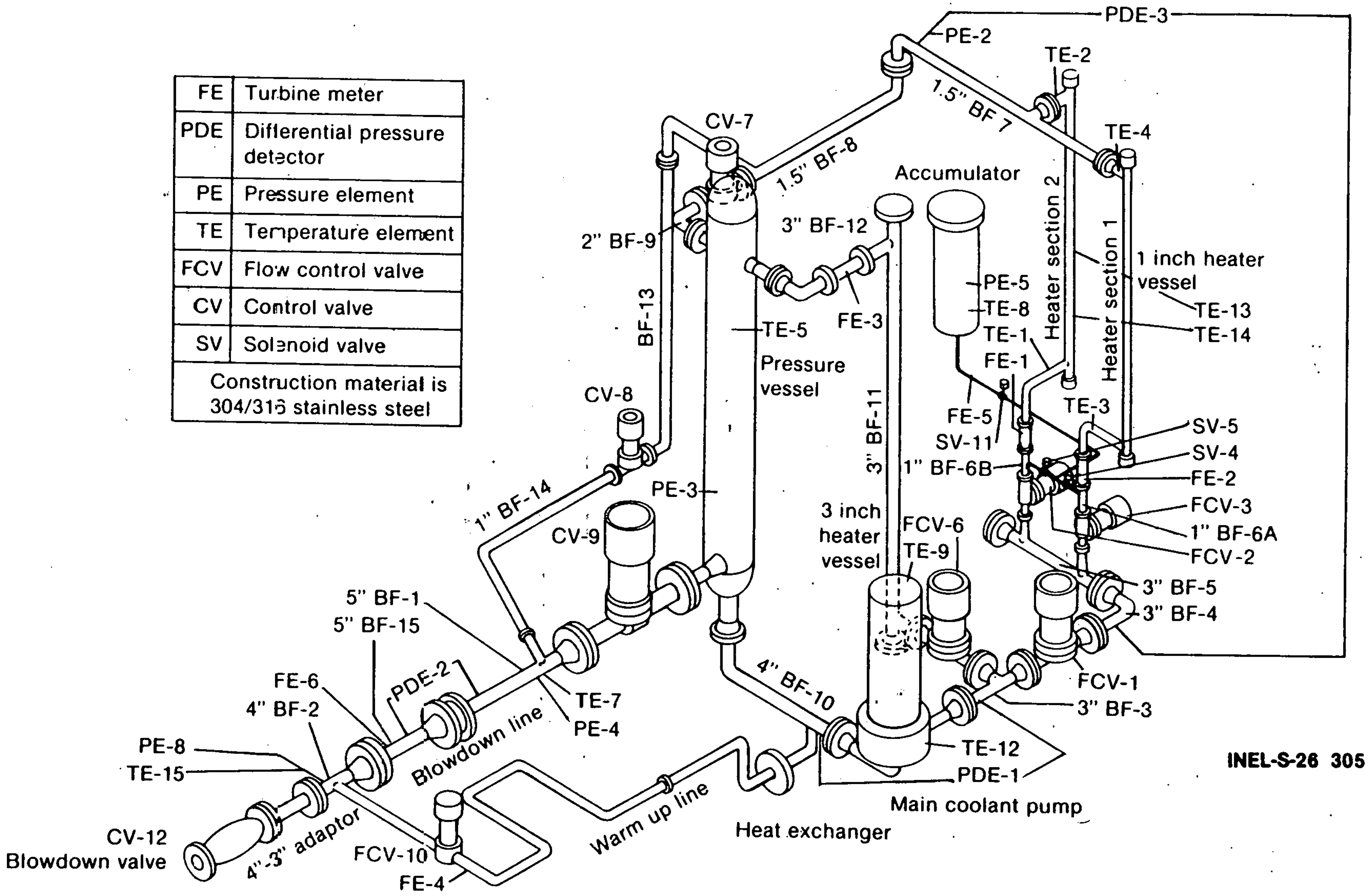


Fig. H.2-2. Blowdown Loop

The major test assembly components include a pressure vessel, a recirculation pump, three electrical heater rod vessels, a test section, and a coolant injection system. Remotely controlled valves are installed in the system to allow flexibility in flow path selection.

Full Area Steady State Loop

The Full-Area Steady State (FAST) loop is a high-pressure 7.72 MPa (1120 psi), high-temperature 552 K (535 F), high-mass flux 302 l/s (4800 gpm) subcooled-water, steady-state test system. A pressurizer, the pump input energy, and a heat exchanger are the primary means used to control pressure and temperature. The unique configuration of the system allows single-phase, continuous flow in either direction through the loop. Three test sections are installed, one horizontal and two vertical. One vertical section is 5.5 m (18 ft) long by 45 cm (18 in.) in diameter, the other is 5.5 m long by 15 cm diam. The horizontal test section can accommodate instruments in a Sweepolet-type insertion port. The horizontal test section is 2.44 m (8 ft) long by 35 cm (14 in.) diam. The system was fabricated from 304 stainless steel throughout, and the volume is 1.72 m^3 (405 gal). The FAST loop test assembly is illustrated in Fig. H.2-3.

Other Facilities

The test loops listed in Table H.2-1 are essentially specialized facilities designed to perform specific, separate-effects type tests. There are also at EG&G some integrated test facilities designed to investigate effects on integrated systems of thermal/hydrualic failures. One of these is the Semiscale Mod-2A system shown in Fig. H.2-4.

The Mod-2A system has a simulated reactor vessel with up to 25 electrically-heated Pressurized Water Reactor (PWR) "fuel rods". Two circulating loops simulate a broken loop and an intact loop of a PWR reactor that has suffered a pipe break [loss-of-coolant accident (LOCA)].

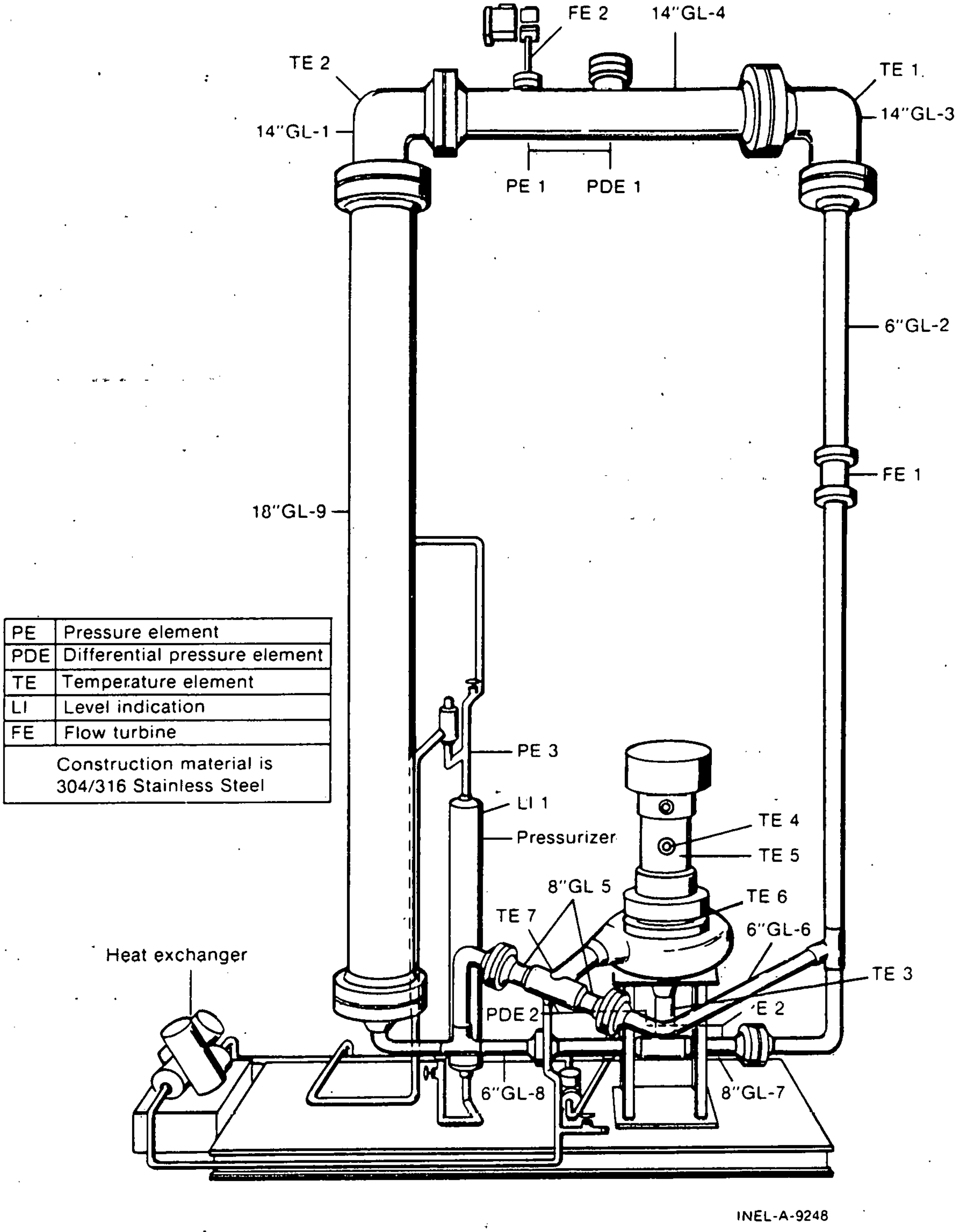


Fig. H.2-3. Full Area Steady State Loop

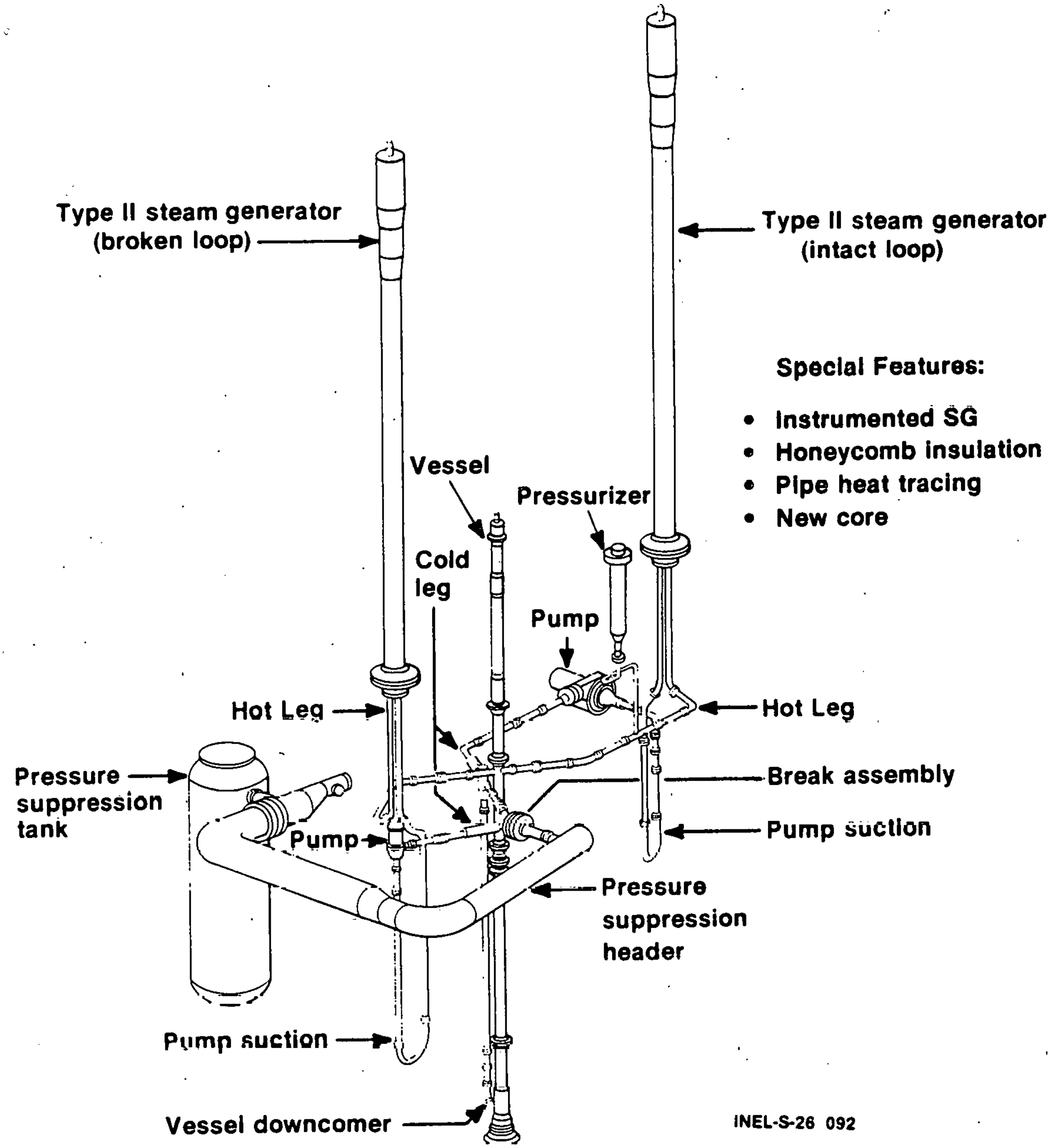


Fig. H.2-4. Semiscale Mod-2A System

TABLE H.2-3
SEMISCALE MOD-2A OPERATING CONDITIONS

Characteristic	Requirement	
Design Pressure	2500 psig	17.1 MPa
Design Temperature	650°F	350°C
Steady State Conditions		
Pressure	2250 psig	15.4 MPa
Inlet Temperature	553°F	390°C
Outlet Temperature	617°F	325°C
Coolant Flowrate	210 gpm	13.2 l/s
Total Power	2 MW	
Transient	Simultaneous rapid fluid thermal transients from 324° to 65°C and rapid pressure transient from 15.4 to 0.1 MPa without damage.	

It has emergency core coolant systems (ECCS) modeled after a large PWR. The vessel will hold a 12 foot long PWR 25 fuel rod bundle, that is electrically heated. The design specifications are listed in Table H.2-3. It is a highly flexible system in that it can be configured to simulate a variety of events. It could be used in FW/B/S non-nuclear testing to investigate similar kinds of failure scenarios.

H.3 HELIUM TEST FACILITIES

Air Flow Test Facility

Helium flow tests can be simulated at very low cost in General Atomic Company's air flow test facility (see Fig. H.3-1 for a schematic). It is housed in an enclosed 2200-sq-ft building with a 3-ton bridge crane; test models as high as 10 m (33 ft) can be accommodated. Larger models, up to 23 m (75 ft), can be installed in an adjacent tower using two cranes with a capacity of 15 tons. Both buildings offer easy access for special handling devices.

Three blowers of 185 kW (250 hp) each comprise the primary air-moving system. These blowers can be operated either in parallel to provide air flow rates up to 19 m^3/s (40,000 SCFM) with a 17 kPa (2.5 psi) pressure rise or in series to provide 5.6 m^3/s (12,000 SCFM) with a 34 kPa (5 psi) pressure rise. A single 670 kW (900-hp) blower can provide 21 m^3/s (45,000 SCFM) with a 34 kPa (5-psi) pressure rise. Several low-capacity air-moving sources are also available. A 90-kW heater bank combined with a 2.8 m^3/s (6000-cu-ft/min) air blower is available for applications requiring a heated air source.

Instrumentation includes many types and sizes of flowmeters, thermocouple reference junctions, hot-wire anemometry gear, acoustic and vibration-measuring equipment, helium-gas-concentration measuring equipment, and standard and specialty pressure probes. Accurate pressure probe calibrations are accomplished by means of an optically aligned wind tunnel and alignment bed.

A digital computer is used to control pressure and temperature high-speed data collection (400 data points simultaneously) and to perform small-scale data reduction.

HOT GAS TEST FACILITY

The \$4 M Hot Gas Test Facility at General Atomic Co. is housed in an enclosed 8000-sq-ft building with a 15-ton overhead bridge crane (see Fig. H.3-2 for a schematic). This facility, designed initially to test main helium circulators for HTGR's, consists of two closed loops:

1. A non-condensing steam loop powered by an electric motor driven 6 MW (8000 horsepower) multi-stage centrifugal compressor circulating 22.7 kg/s (50 lbs/s) steam at 385°C (725°F) and 0.9 MPa (126 psia) discharge pressure. The compression pressure ratio is 2.2 allowing for sonic velocity tests or for driving steam turbines with up to 163 kJ/kg (70 btu/lb) enthalpy drop. This loop can also be operatd with helium as the flowing medium to test components in an auxiliary test section. Temperatures and pressures are similar to those with steam.

2. Helium loop. This is a closed loop with a design capacity of 49 kg/sec (107 lb/sec) helium flow at 330°C (625°F), inlet pressure of 1 MPa (147 psia) and a pressure rise of 34 KPa (5 psi). The flow in this loop is generated by the helium circulator driven by the above described steam loop. This loop is housed inside a 4 m (156in.) diameter, 5 m (16 ft) tall, pressure vessel rated at 1.4 MPa (200 psig). This loop could be easily adapted to circulate other gases. The main helium vessel has two flanges which can be used to pipe the hot gas outside of the vessel to another test vessel and back. This can be combined with an afterheater and precooler allowing for the testing at temperatures up to 870°C (1600°F), pressure of 5.5 MPa (800 psia), a flow rate of 32 kg/sec (70 lb/sec), and a pressure drop of 0.1 MPa (15 psia). Typical velocities in this loop could go as high as 300 m/s (1000 ft/sec).

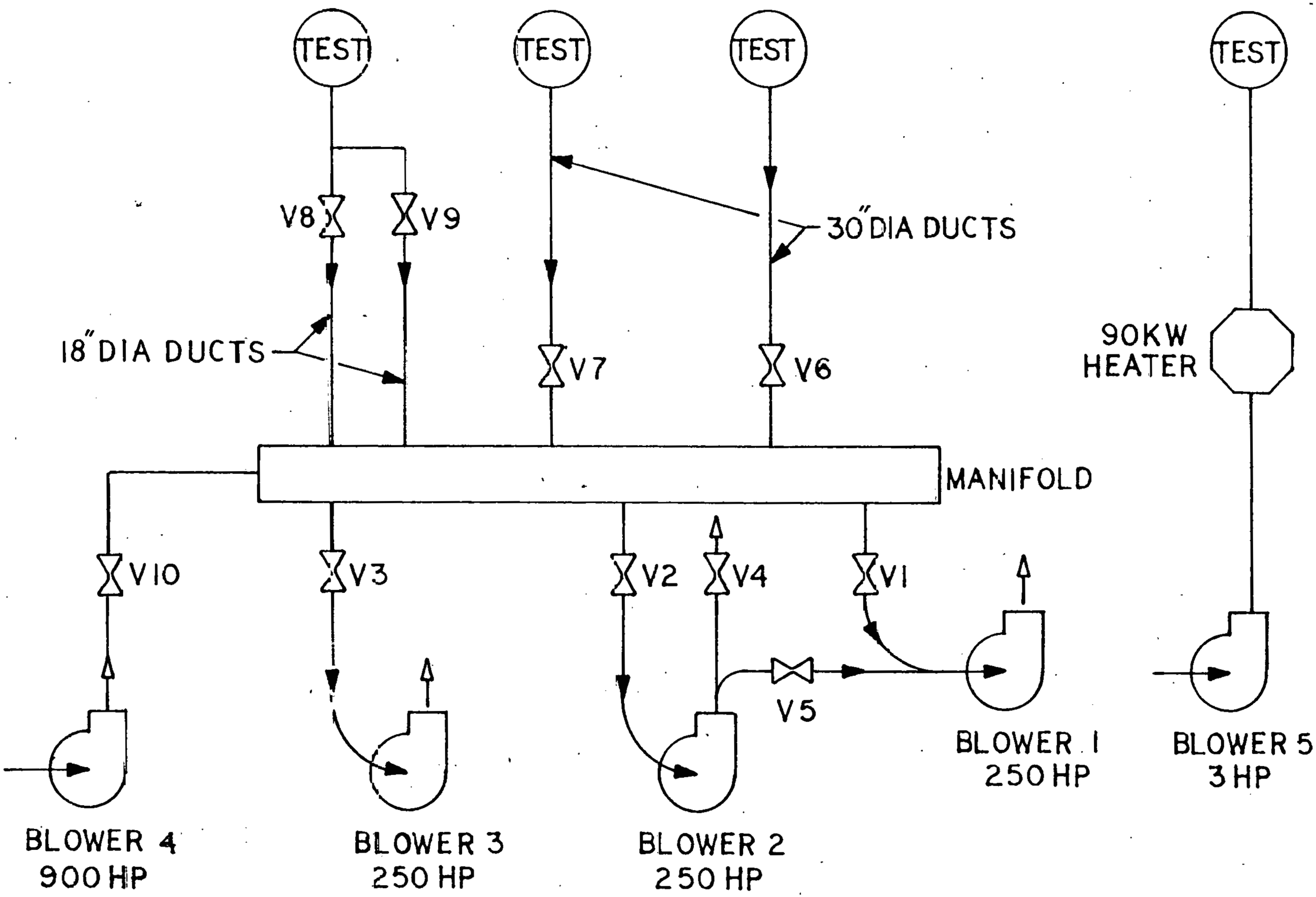


Fig. H.3-1. Schematic of Air Flow Test Facility

V_1 = Speed control valve

V_2 = Bypass valve

V_3 = Temperature control valve

V_4 = Secondary gas flow control valve

M_1 = Motor 8000 hp

C_1 = Primary gas compressor

C_2 = Secondary gas compressor

T_1 = Turbine

E_1 = Primary gas cooler

E_2 = Secondary gas cooler

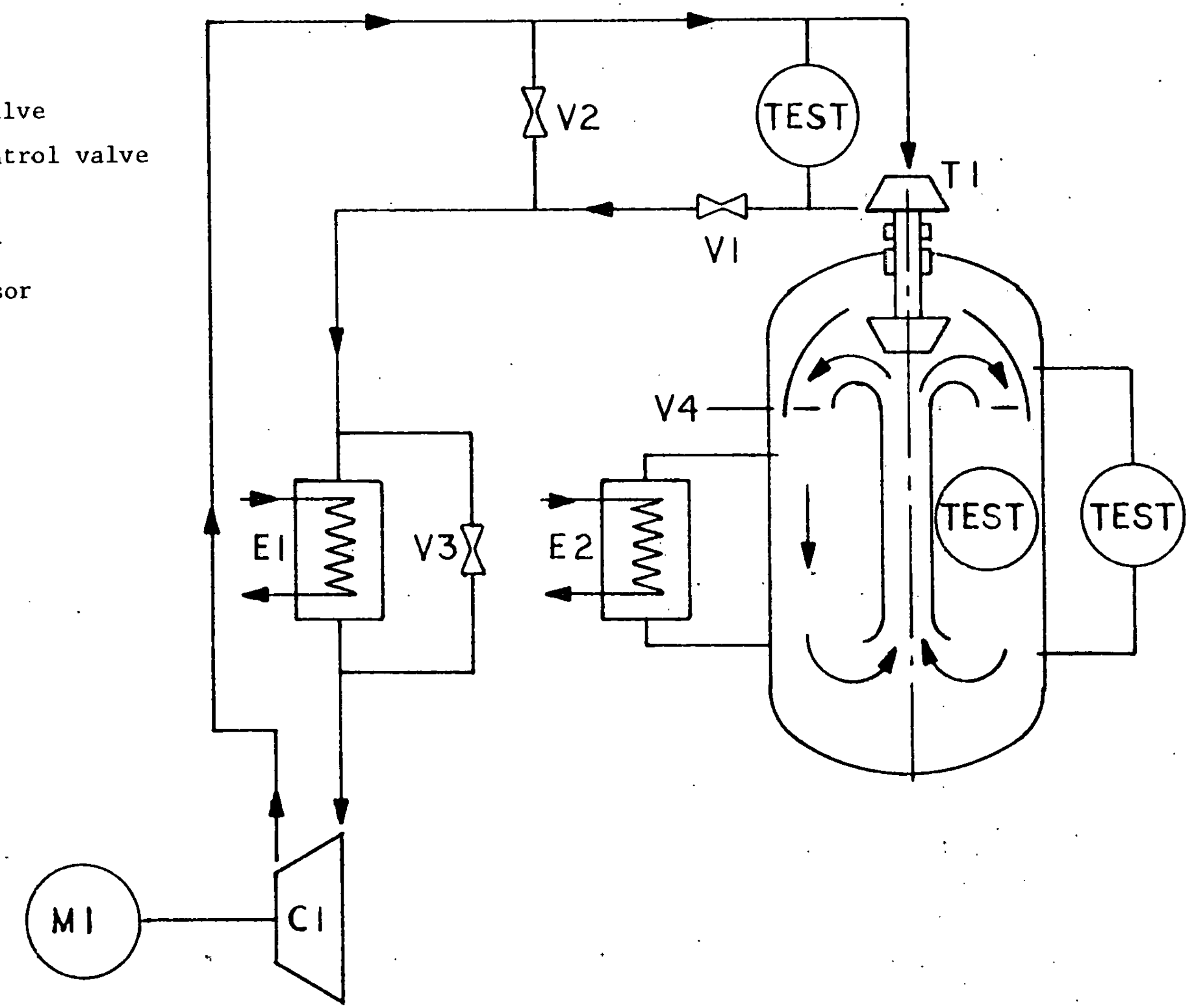


Fig. H.3-2. Hot Gas Test Facility Schematic

This is a highly automated facility with elaborate data acquisition system suitable for operational flexibility with respect to flow, temperature and pressure. Instrumentation and controls for the loops consist of nuclear grade Foxboro Spec 200 components and systems. Interlock and protection circuits are provided for safe operation and shutdown of the loops. Existing test instrumentation is available to monitor multiple channels of temperature, pressure and flow and for rotating parameters such as speed, vibration and wobble.

Helium Flow/Vibration Test Rig

The helium flow/vibration test rig at General Atomic Co. consists of a closed stainless steel loop with a test chamber in which helium flows at specified conditions. The test loop is located in the west pit of the test bay area of the GA hot gas test facility with full access to extensive instrumentation and support equipment. A compressor circulates helium through the test chamber and then through a heat exchanger where the excess heat is removed. A bypass line around the test chamber permits control of the mass flow rate through the test specimen. Temperature control is achieved with a bypass line around the heat exchanger.

The instrumentation of the loop includes temperatures and pressure gauges and flowmeters. Mass flow rates of helium are determined with orifice flowmeters located at the outlet of the compressor and at the outlet of the test chamber. The operation and control of the loop is entirely manual.

The test vessel consists of a 0.2 m (8 in.) by 3 m (10 ft) long tube which is joined to the loop by bolted flanges. However, by using these flanges any other vessel sizes or shapes could be easily adapted. Currently the test conditions are temperatures up to 200°C (400°F), pressures to 5.9 MPa (850 psia), and a flow of 0.08 m^3/s (180 acfm).

Core Flow Test Loop

General Atomic has participated in the design and construction of the $40 M Core Flow Test Loop located at ORNL. This facility could be used for helium-cooled blanket tests and will be included in the test facility evaluations done in TPE II. The facility is capable of the conditions shown on Table H.3-1.

H.4 LIQUID METAL TEST FACILITIES

To support the Liquid Metal Fast Breeder Reactor test effort, a Filling, Storage and Remelt (FS&R) Facility was constructed in the EG&G ETR facility. This facility is comprised of an experiment assembly in a shielded test cell, a sodium circulating and purification system, a vacuum system, and a large oven for heating test components to 300°C (500°F). (The test space is approximately 50 cm (19 in.) diameter by 90 m (27 ft) long). The FS&R is used for out-of-pile verification testing. To accomplish this, over 250 channels of data acquisition tie-in's are available in the FS&R as well as a tie-in to the SLSF helium system. The facility is shown in Fig. H.4-1.

In addition, the FS&R is equipped with another small test loop, called the Component Test Loop for testing components in a high temperature liquid metal environment above 600°C (1000°F).

The initial testing of blanket modules using liquid lithium could be simulated using liquid sodium. The sodium heat transfer characteristics could be scaled for lithium, thus permitting testing with a minimum of modifications to the FS&R facility.

While the blanket modules would not use sodium, similar types of systems would be required to support the assembly and out-of-pile testing of FW/B/S components. The installed sodium charging equipment is skid mounted to permit removal and replacement by other systems. The oven design would be specific to the module or components being tested.

TABLE H.3-1
MAIN CHARACTERISTICS OF THE CORE FLOW TEST LOOP (CFTL)

Design Pressure:	11.8 MPa (1715 psia)	
Operating Pressure:	Ambient to 10.6 MPa (1540 psia)	
Power:	0 to 4 MW, controlled by 13 independent power supplies	
Temperature:	260 to 600°C. Atemperation flow arrangement allows stainless steel melting (approximately 1350°C) in the test section.	
Transients:	Full power to zero power in	1 second
	Full flow to zero flow in	1 second
	Full pressure to approximately ambient less than	1 minute
	All transients fully program controlled	
Helium Circulators:	Centrifugal type with gas lubricated bearings, hermetically sealed	
Working Media:	Designed for helium with impurity control	
Flow Rate:	0 to 3.2 kg/s (circulators in series); to 9.6 kg/s (circulators in parallel)	
Flow Measurement:	Wide range, high accuracy vortex shedding flowmeters	
Data Acquisition:	High speed (10 kHz) 640 channels (expandable) computer controlled	

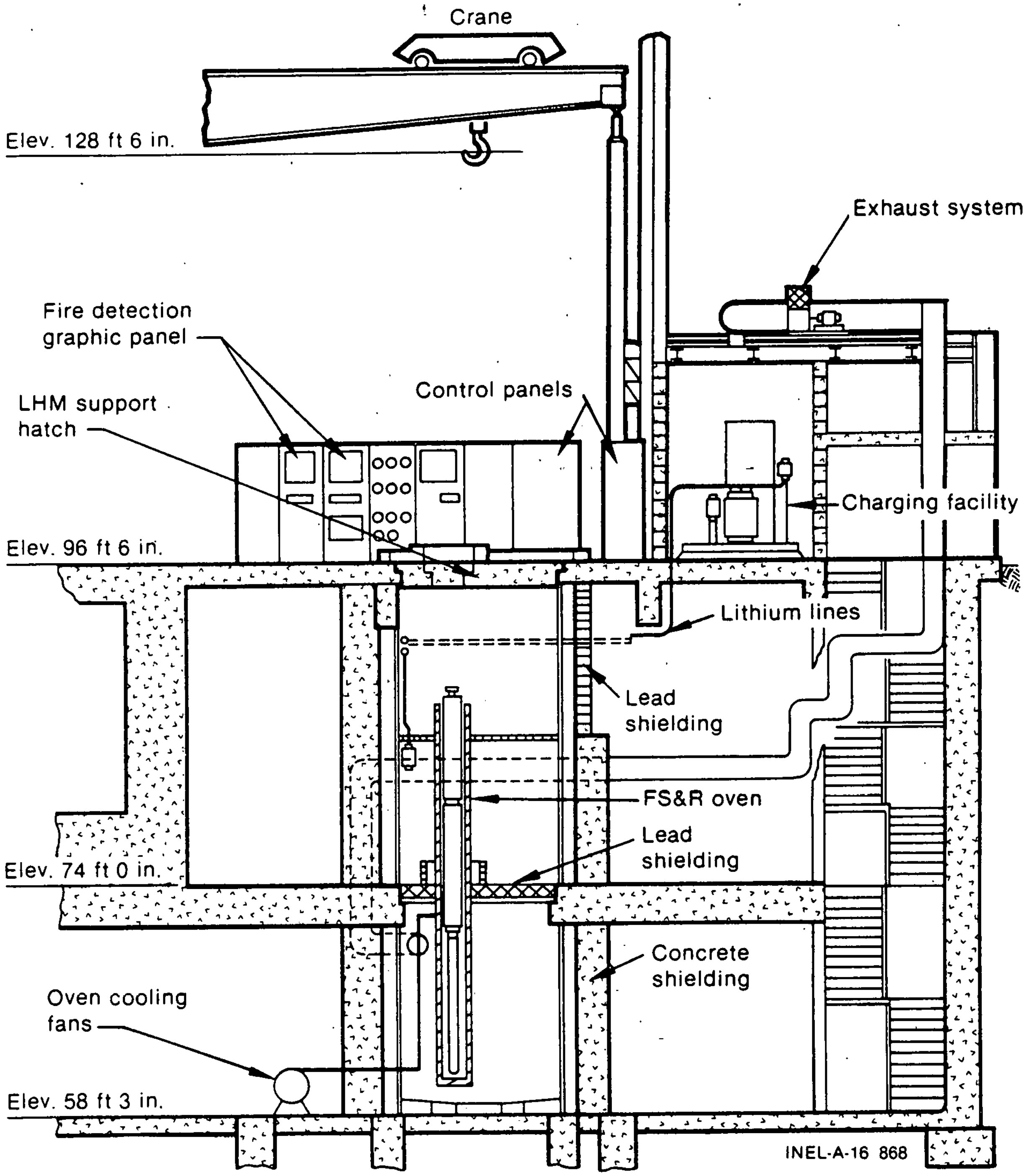


Fig. H.4-1. Liquid metal Filling, Storage and Remelt (FS&R) Facility.

H.5 FISSION TEST REACTORS

Engineering Test Reactor

As indicated in Section C, it is important that nearly full-scale blanket modules to be used in systems testing. To test blankets as now perceived will require large volumes with high bulk heating, surface heating, and neutron fluxes. No present test facilities can provide all three environments simultaneously. However, the Engineering Test Reactor (ETR) at EG&G has the flexibility, test volume, radiation environment, and availability required to conduct a meaningful first wall/blanket systems test program. Several reactors have partial capabilities for scaled systems tests and component testing, but none appear to be as suitable as the ETR for testing of this kind, as set forth in Table H.5-1.

The ETR shown in Fig. H.5-1 is unique in that it contains, within the core region, a large number of experimental facilities. This point is illustrated in Fig. H.5-2 which shows a cross section of the reactor through the pressure vessel. The reactor core occupies a matrix of 100 3 x 3-in. squares and, of these, the experimental facilities occupy 31 positions. As shown in Fig. H.5-2, these facilities consist of three 6 x 6-in. experimental holes, four 3 x 3-in. holes, one 6 x 9-in. facility, and one 9 x 9-in. experimental region. Each of these facilities is surrounded by fuel and all except two of the 3 x 3-in. facilities are through holes. This means that an experiment enclosed in a pipe may enter the pressure vessel above the core, go completely through one of the experimental holes, come out under the grid plate which supports the core, and leave the pressure vessel at the bottom. Thus, it is possible to arrange experiments in the form of self-contained loops that pass completely through the reactor core region and that may contain components such as flowmeters, pumps, heat exchangers, etc. which are isolated from the reactor equipment and instrumentation. The design of these facilities permits one to examine the effects of high neutron and gamma fluxes on nuclear fuels, materials, components, and complete sections of devices which will operate in high radiation fields.

TABLE H.5-1
TEST REACTOR COMPARISONS

Reactor	Availability	Present Maximum Test Volume (liters)	Neutron Flux Density ($cm^{-2}sec^{-1}$)	Features
MITR	Possibly	2.1	3×10^{13} Thermal 1×10^{14} Fast	In-core thimbles
ORR	Immediate	3.6	1.5×10^{14} Thermal 4.5×10^{14} Fast	Replacement of fuel element with experiment
HFIR	Partially	6.8	2.8×10^{15} Thermal 1.3×10^{15} Fast	
ETR	Immediate	48 (451[a])	2.5×10^{14} Thermal 4.0×10^{14} Fast	Versatility very high
EBR-II	Immediate	3.0	2.0×10^{15} Fast	
FFTF	Immediate	8.6	4.6×10^{15} Fast	Any core fuel or reflector position

[a]With core reconfiguration.

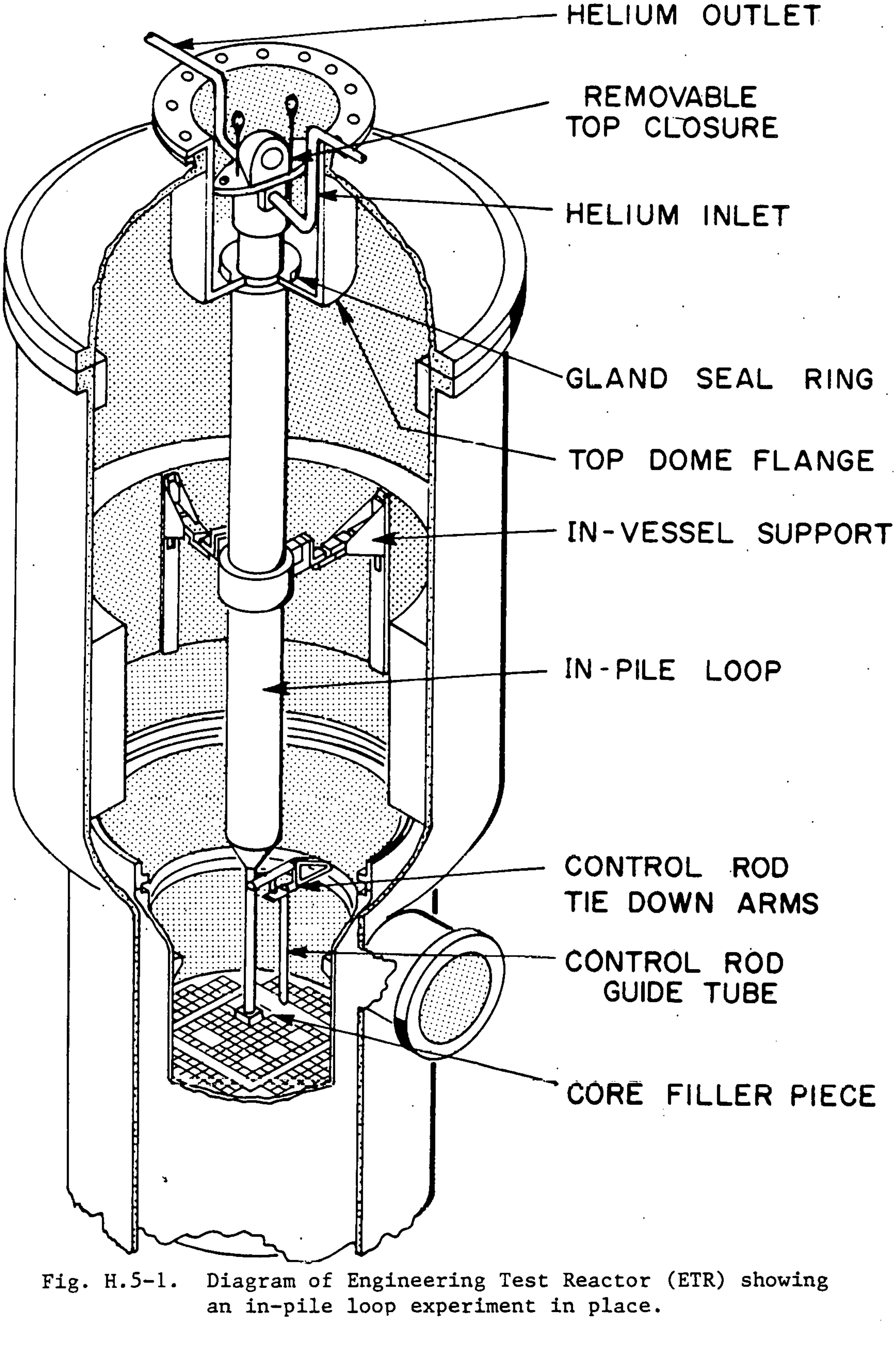


Fig. H.5-1. Diagram of Engineering Test Reactor (ETR) showing an in-pile loop experiment in place.

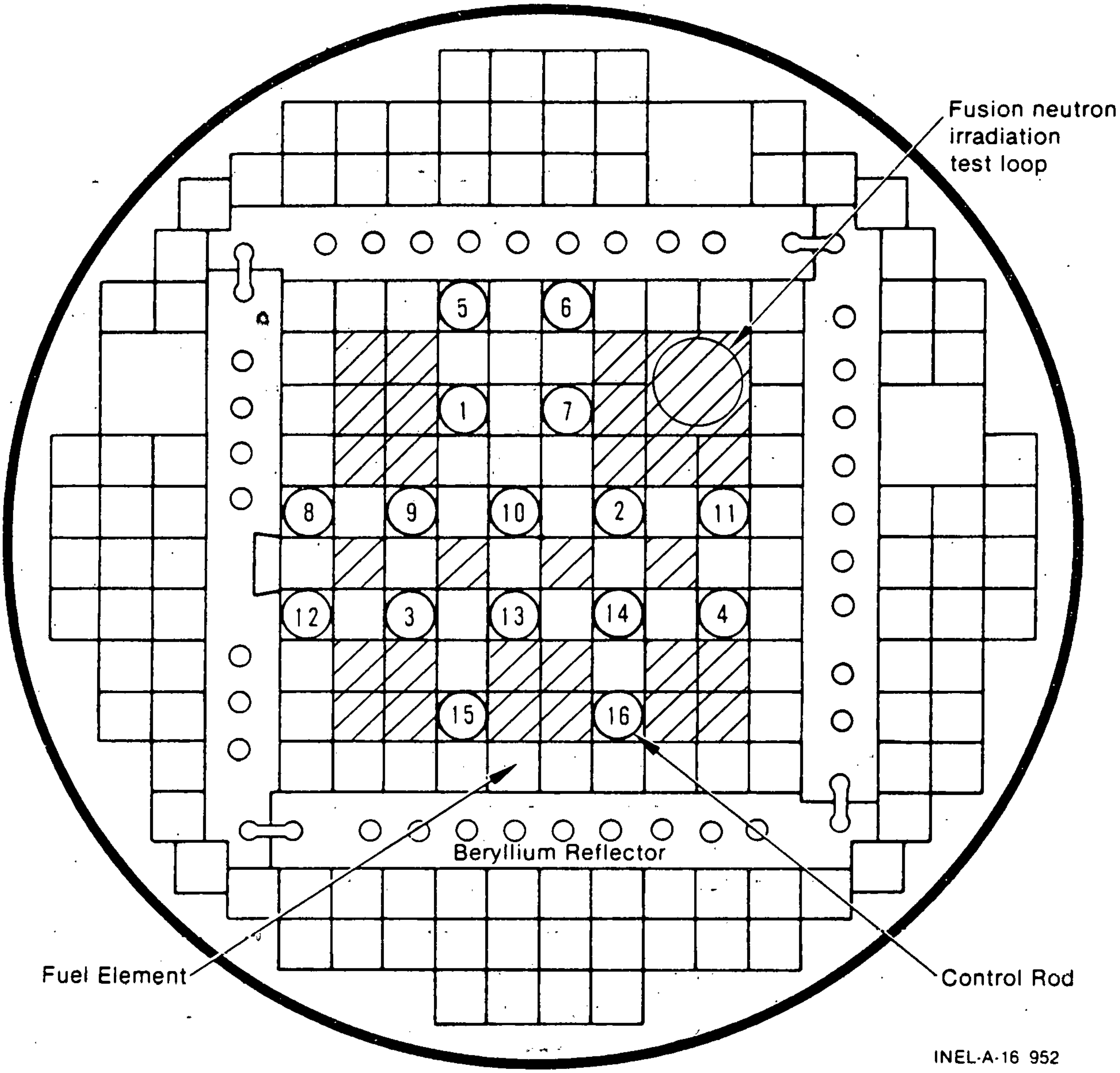


Fig. H.5-2. ETR core configuration cyindrical module.

Outside the core region and surrounding it is a slab beryllium reflector that contains additional experimental holes into which capsules containing test materials may be placed. Surrounding the slab beryllium reflector is a region containing aluminum reflector pieces. Each reflector piece contains a hole into which samples may be placed, and each piece is removable so that large experimental regions may be made available in the aluminum reflector region. Two 6 x 6-in. and six 3 x 3-in. through facilities similar to those in the core are also contained in the aluminum reflector region.

The radiation environment of the ETR can be tailored to provide a neutron spectrum which, except for the 14 MeV spike, is similar to the neutron spectrum anticipated in the first wall of a fusion reactor. This comparison for a FW/B/S module employing ^3He to simulate surface heating in ETR is shown in Fig. H.5-3. The high flux (see Table H.5-1) will allow accelerated testing of FW/B/S hardware.

The core configuration of the ETR is highly flexible. By moving fuel plate assemblies and control rods to different locations within the 100 element matrix, it is possible to accommodate even larger test items. This is illustrated in Fig. H.5-4 which shows a conceptual core arrangement for doing FW/B/S testing.

The ETR is also readily available. Present programs planned for it will be completed by 1982. It has the potential to become a dedicated facility, although economic advantages accrue from running "piggy-back" experiments simultaneously.

Supporting Subsystems. The experimental air system adjacent to the reactor building provides a flow rate of up to 14 kg/sec at pressures of 350 to 2200 kPa. In addition, up to 11 kg/sec airflow can be heated at temperatures in the range of 260 to 650°C.

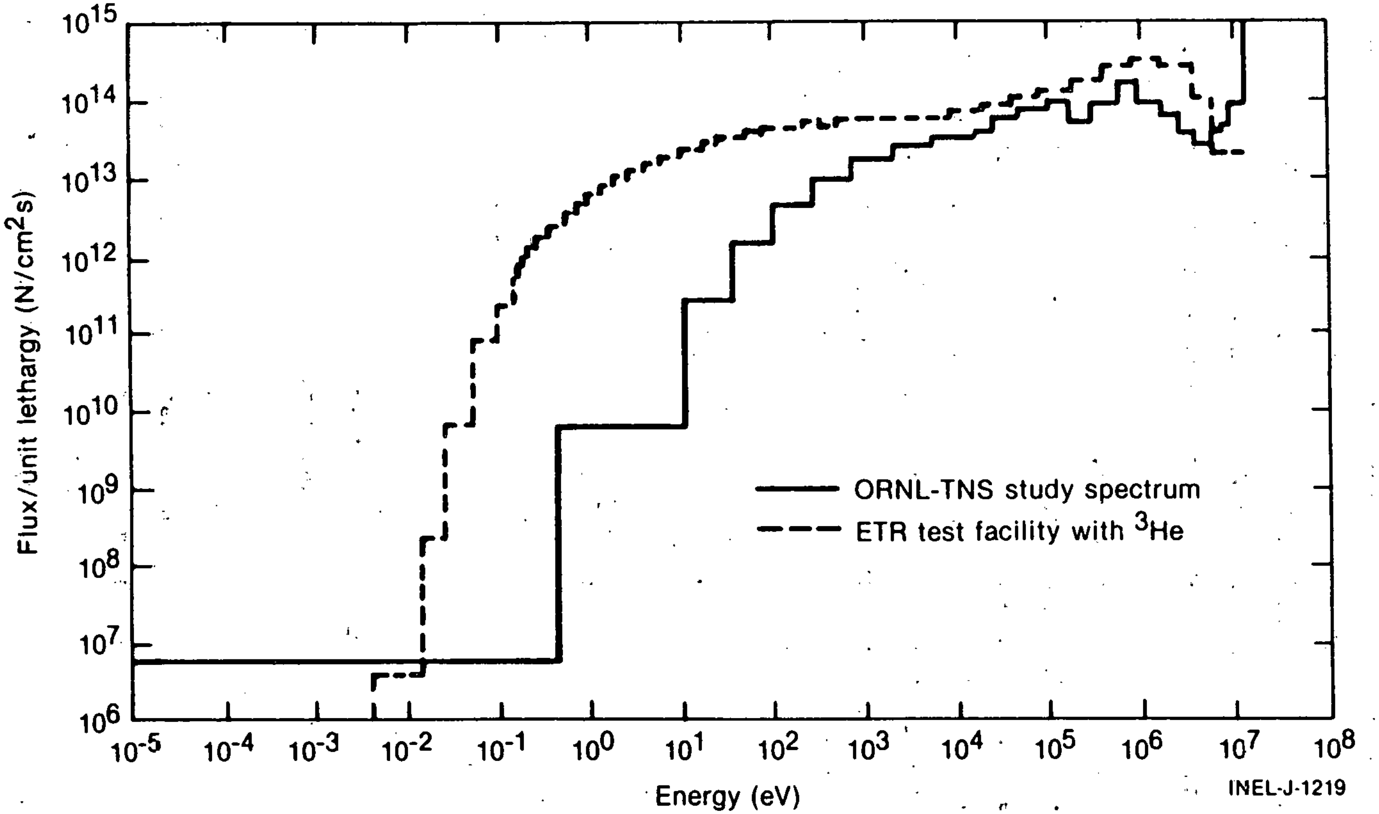


Fig. H.5-3. Comparison of fusion first wall neutron spectrum with that in a FW/B/S module with ^{3}He surface heating in ETR.

The helium circulation system now employed in the SLSF has a design capability of 0.83 kg/sec at 1700 kPa and at a maximum temperature of 600°C.

Liquid metal handling capability in the ETR is provided by the Falling, Storage, and Remelt (FS&R) facility. This system provides all the capabilities necessary to operate a closed liquid metal loop in the ETR. Details on the system can be found in Section H.4.

Summary. The ETR has the flexibility, radiation environment, and experimental volumes required for successful nuclear experiments on most realistic first wall/blanket concepts. Liquid metal loops have been operated successfully in the reactor since 1976. Near-term testing could be done with very little extension to the existing capability for SLSF testing.

H.6 OTHER FACILITIES

Hot Cell Facilities

Hot Cell facilities at GA and EG&G include equipment and expertise giving a range of capabilities. Special capabilities for nondestructive evaluation include fuel rod gamma scanning (gross and isotopic), neutron radiography, pulsed eddy current scanning, automated dimensional measurements and channel spacing on fuel rod bundles. Destructive evaluations include gas analysis for fuel rods (fill gas, metallography and microhardness), remotized scanning electron microscopy (SEM), burnup analysis of both fuel and cladding for fuel rods, and replication of ballooned rods and artifacts on other materials. Remote handling of fuel rods extends to assembly as well as disassembly.

In addition to the hot cell facilities mentioned above which are operated by EG&G Idaho, ANL-West operates the Hot Fuel Examination Facility - North (HFEF-N) which is the largest alpha-gamma hot cell facility in the

United States. This facility has handled irradiated fuel and materials experiments from EBR-II and can accommodate large experiments from other facilities. The HFEF-N is divided into an air atmosphere decontamination cell of 56 m^2 area and an adjacent main argon atmosphere cell having an area of 195 m^2. An extensive array of equipment is available in the HFEF-N for both destructive and nondestructive examination.

High Temperature Furnaces

EG&G has two high temperature furnaces with unique capabilities which could be very useful in non-nuclear testing of FW/B/S components. These furnaces are commercial units manufactured by Richard D. Brew Co. of Concord, NH. Each consists of a vacuum chamber, vacuum pumps (mechanical and diffusion), power transformers, and associated controls. These are two systems, commonly referred to as the big Brew and the small Brew. Each vacuum chamber includes heater units made of tungsten mesh, and can be used with inert gases. Further specifications are as follows:

	Small Brew	Large Brew
Useable Work Area	2 in. x 8 in.	6 in. diam x 34 in. (four independently controlled heat zones)
Rated Temperature		
Vacuum	3000°C	3000°C
Helium (ultra high purity)	2500-2700°C	2500-2700°C
Argon (high purity)	2000-2300°C	2000°C-2300°C
Hydrogen	N/A	2000°C

Each furnace has viewing ports in the chambers. Optical pyrometers are available for use with the systems. The ultimate vacuum obtainable in the furnace chamber is approximately 2 x 10^{-6} Torr.

Description of Existing Facility

General Atomic and EG&G, Idaho, have many facilities for conducting thermal-hydraulic and thermomechanical testing of components and assemblies. One which has great capability and flexibility is a furnace at EG&G manufactured by Richard D. Brew and Company, known locally as the "Big Brew," which is described in Table 2-4. This large, high-temperature furnace has four separately controllable heat zones, each of which is 7 in. in diameter and 7.75 in. high. The usable work space inside the furnace is approximately 7 in. in diameter and 34 in. high. There work access ports 8 in. in diameter at each end of the furnace and four viewing and instrument lead windows 3 in. in diameter at each zone. The chamber itself can be opened via a large, hinged, half-cylinder door that extends the entire length of the chamber to allow easy access to internal components or test assemblies.

Heating is by tungsten mesh. The maximum rated temperature of the furnace is dependent on the atmosphere in the test chamber, as follows:

1. 3000°C in vacuum (operating vacuum 5×10^{-5} Torr)
2. 2500 - 2700°C in high-purity helium
3. 2000 - 2300°C in high-purity argon
4. 2000°C in hydrogen

The temperature uniformity of a heating element is within ±10°C as measured by a disappearing filament-type pyrometer.

Figure 2-2 is a schematic drawing of the "Big Brew" furnace with the proposed experiment in place. Each of the heat zones is separately controllable. Controls are manual, one or more zones may be tied to a single preprogrammed temperature cycle implemented by a curve tracking control. Each heat zone has separately controlled, water-cooled copper heat sinks that surround the heat zone just outside the heating elements. The heat sinks are isolated from the heating elements by a five layer, tungsten radiation shield.

TABLE 2-4
"BIG BREW" BLANKET TEST FACILITY SPECIFICATIONS

1. Heating Capability

Power: 530 kW

Test Volume: 18 cm diam x 79 cm long

Equivalent Heat Flux: 120 W/cm^2

Equivalent Volumetric Heat Source: 26 W/cm^3

2. Engineering Data

A. Heating Element Size: 18 cm diam x 20 cm high

B. Maximum Temperature: 3000 °C in vacuum, 2700 °C in helium

C. Temperature Uniformity: ±10 °C

D. Heating Element Material: Tungsten Mesh

E. Heat Shield Material: Tungsten Sheet

F. Operating Vacuum: 5×10^{-5} Torr

G. Ultimate Vacuum: 1×10^{-6} Torr

3. Facility Requirements

A. Electrical: 530 kVA at 480 V/3 phase/60 Hz

B. Water: 3.5 ℓ/s (54 gpm) at 25 °C maximum and 0.3 MPa (50 psig) minimum

C. Air: 0.3 MPa (50 psig) minimum

GACP 11-118

Availability

This furnace, shown in Fig. 2-3, is presently located in the Central Facilities area, Building 688, at the INEL. Installation is essentially complete. There remains only final checkout and fine tuning to make the system fully operational. All necessary hardware is available. Facility power, air, and water have been connected. There are presently no other major programs that require this furnace. It will thus be available for FW/B/S testing essentially full time. Furthermore, there are shop facilities immediately adjacent to the furnace (next room) for fabricating parts, assembly, examination, and repairs as needed.

Description of New Components to be Added

The test facility will consist of the existing "Big Brew" furnace, which serves as the test bed and primary heat source, and a test assembly that will be fabricated and mounted in the furnace. The test assembly, shown in Fig. 2-2, is cylindrical, occupying the entire interior working volume of the Brew furnace, with a diameter of 15.2 cm and mounted on top and bottom 20 cm flanges, for a total axial length of 86 cm.

The test assembly itself is made up of two main sections. The upper section, the test-bed module, which is described in Section 2.2.5, actually contains the breeder bed to be tested. The lower section of the test assembly, the test support module, contains the mechanical interfaces needed to support the test module. The most important component in the lower section is a radiant helium heater, which occupies the entire working bore of heat zone 4. The present concept for this heater consists simply of a small diameter (1.3 cm) tube wound on the exterior of a 12.7 cm diameter support cylinder, yielding an effective diameter of 15.2 cm. This heater receives cold helium through a connection in the bottom flange and discharges hot helium into the plenum indicated in Fig. 2-2. Since this helium heater occupies only zone 4, the heat rate to the hot helium stream is independently controllable. In addition, a separate cold helium stream is introduced through the bottom flange and mixes with the hot stream in the plenum before reaching the bottom of the test bed. This allows complete control of the helium flow rate and temperature, via controls on the zone 4 heater, and the hot, cold, and total helium flow rates.

At the bottom of the test assembly are the connections for helium, any required cooling (except for the test bed coolant tube), and all permanent test assembly instrumentation (see Section 2.2.6). In addition, formed bellows are provided to accommodate the axial thermal expansion of the entire assembly when it is at high temperature. All of the flanged connections at the bottom of the assembly include normal elastomer O-rings to make the vacuum and helium seals. At the top flange, the seals can be either water-cooled elastomer O-rings or hard metal. The seal between the test bed module and the lower support section must be made up with a hard metal seal, probably an Inconel ring.

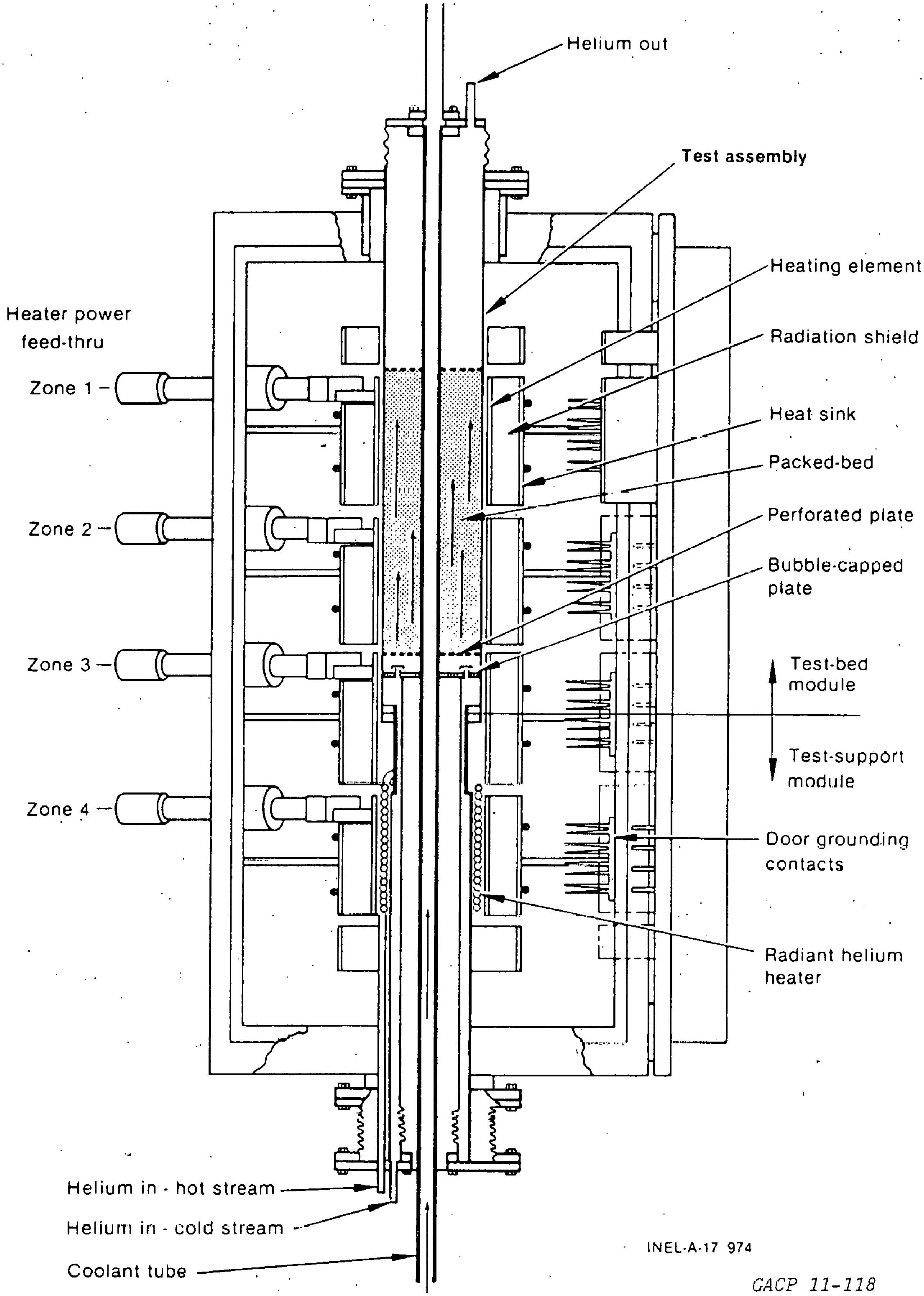
The entire test assembly is designed to operate with a vacuum in the furnace volume and with a helium purge pressure of 0.1 to 0.3 MPa (15 to 50 psig). The helium will be supplied from standard pressure cylinders. The water coolant channel will be designed to operate at simulated blanket conditions up to 300°C and 15 MPa. The material used for construction will be 316 stainless steel.

2.2.5 Description of Test Pieces

For each lithium compound material and each configuration, two identical test modules will be needed, one for the steady-state tests and one for the cyclic tests.

Test Module

For the non-nuclear simulation tests, the test module shown in Fig. 2-2 will be used. This module will be 15 cm (6 in.) in diameter and



GACP 11-118

Fig. 2-2. First Test Facility Schematic

approximately 60 cm (24 in.) in length, constructed of stainless steel. It will be designed to withstand an internal pressure of 0.3 MPa (50 psig) in the bed area up to and 15 MPa (2250 psig) in the coolant tube.

The lithium compound bed itself, which is approximately 43 cm (17 in.) in length, will occupy the volume enclosed by the upper two heat zones of the Brew furnace. The bed might be prepared in several ways, in the form of pressed powder, packed beds of microspheres or micropellets, or sintered pellets. Each of these has its own advantages and disadvantages. GA is experienced in the fabrication of the various lithium compounds in the various forms. Some examples are shown in Fig. 2-4. The decision on which approach to use for the first test will be made jointly among GA, EG&G, and ANL at the initiation of the Phase I program.

Whatever approach is used, the bed will then be packed and partially sintered in place between two perforated plates. Below the lower plate will be a second plate with flow holes covered by bubble caps (similar to those used in fluidized beds) to prevent bed debris from falling through into other parts of the system. A central stainless steel coolant tube is welded to the lower plate and is part of the module. This tube extends out of both the top and bottom of the test assembly to allow connections to be made. The test bed module will be bolted to the test support module (see Section 2.2.4) and a seal will be made with a metal ring. The top of the test bed module will mount on the upper mounting plate of the Brew furnace.

Note that this test module configuration is representative of the "solid breeder canister with pressurized coolant tubes" blanket concept (the STARFIRE blanket concept) and should allow fairly realistic testing of the critical thermal-hydraulic features of this design. The design proposed for the test module to be used in the first TPE II Phase I experiment is also a direct precursor of test modules that would be used for in-pile nuclear testing of the blanket design later in TPE II. The nuclear tests will use a similar test module and will add volumetric nuclear heating, tritium production, and radiation damage effects to the simulation.