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THE HIGH PERFORMANCE RESEARCH REACTOR FUEL DEVELOPMENT HYDRAULIC TEST LOOP

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ABSTRACT

The Convert branch of the National Nuclear Security Administration (NNSA) Global Threat Reduction Initiative (GTRI), is working to develop high uranium density fuels for research reactors. This fuel is intended to enable conversion from highly enriched uranium (HEU) based fuel to low enriched uranium (LEU) based fuel for the high performance research reactors (HPRR). There are five U.S. Reactors that fall under the *HPRR* category including the Massachusetts Institute of Technology Reactor (MITR), National Bureau of Standards Reactor (NBSR), Missouri University Research Reactor (MURR), Advanced Test Reactor (ATR), and High Flux Isotope Reactor (HFIR). Two fuel designs are being developed around a uranium-molybdenum alloy fuel phase, a dispersion and monolithic version. Significant progress has been made recently, with regard to the fuel's microscopic performance, however, additional study of the macroscopic behavior of the fuel elements must be examined before the fuel can be qualified. One critical area of study is the behavior of reactor specific fuel elements under prototypic hydraulic conditions. Although the intent of this project is to conduct a hydraulic study for all of the above reactors, particular attention will be directed toward the ATR due to conditions in which the reactor operates and current available information. This document outlines the current and proposed future progress for the hydraulic test facility as part of the RERTR fuel development program.

1 INTRODUCTION

The Convert branch of the National Nuclear Security Administration (NNSA) Global Threat Reduction Initiative (GTRI), is working to develop high uranium density fuels for research reactors. This fuel is intended to enable conversion from highly enriched uranium (HEU) based fuel to low enriched uranium (LEU) based fuel for the high performance research reactors (HPRR). There are five U.S. Reactors that fall under the HPRR category including the Massachusetts Institute of Technology Reactor (MITR), National Bureau of Standards Reactor (NBSR), Missouri University Research Reactor (MURR), Advanced Test Reactor (ATR), and High Flux Isotope Reactor (HFIR).

This previous research has been centered around several objectives; although the enrichment of Uranium 235 is reduced in the LEU fuel, it is desired to maintain equal reactor performance characteristics while meeting all safety requirements. Two fuel designs are being developed around a uranium-molybdenum alloy fuel phase, a dispersion and monolithic version.

Significant progress has been made recently, with regard to the fuel's microscopic performance, however, additional study of the macroscopic behavior of the fuel elements must be examined before the fuel can be fully implemented. One critical area of study is the behavior of reactor specific fuel elements under prototypic thermal hydraulic conditions.

Although the intent of this project is to conduct a thermal hydraulic study for all of the above reactors, particular attention will be directed toward the ATR due to conditions in which the reactor operates under and current available information.

2 RESEARCH OBJECTIVES

2.1 Stage 1

The first stage of the study will consist of examining the current safety analysis methodologies used for the five U.S. High Performance Research Reactors (MIT, NBSR, MURR, ATR, and HFIR) to establish how experimental studies and analytical studies have been historically applied to resolve hydro-mechanical and thermal-mechanical interactions. This study would be used to determine where a change in fuel form might require new experiments or analysis. It is also key to evaluate how the new fuel design responds to the hydraulic loads (i.e. gross plate deformation, plate tip deflections, vibration, etc.) as well as to thermal loads (i.e. thermal expansion, buckling threshold, etc.) for all five reactors.

2.2 Stage 2

The second stage of this study will consist of the design and construction of a test facility capable of examining a wide range of HPRR elements required to develop a data base for the certification of the newly developed LEU fuel elements. The test facility should be able to provide all the information the previous High Temperature Hydraulic Test Loop [1] collected including high fidelity vibration measurements. The facility is expected to be able to handle fuel elements containing depleted uranium and potentially LEU fuel.

3 BACKGROUND

3.3 ATR Description

The ATR is designed as an experimental irradiation facility and therefore provides for the insertion of numerous experiments into the core. The ATR which is located at the Idaho National Laboratory (INL) is a 250 MW_{th} high flux test reactor designed to study the effects of radiation on samples of reactor structural materials, fuels, and poisons. Construction began in November of 1961 and was completed in 1965. Fuel loading commenced in 1967 and core testing completed in 1969. Full power operation began in August 1969 and the first experiment operating cycle began in December 1969 [2].

The core contains forty fuel elements arranged in a serpentine pattern to form nine flux trap regions, this can be seen in Figure 3.1 below. Each fuel element (Figure 3.2) forms a 45 degree sector of a right circular cylinder and currently consists of nineteen fuel plates with coolant channels on both sides of each plate. The fuel plates are 49.5 in. long with an active fuel length of 48 in. loaded with highly enriched uranium matrix (UAl_x) in an aluminum sandwich plate cladding. Boron is included in certain plates as a burnable poison to minimize radial power peaking and extend the cycle life of the fuel elements.

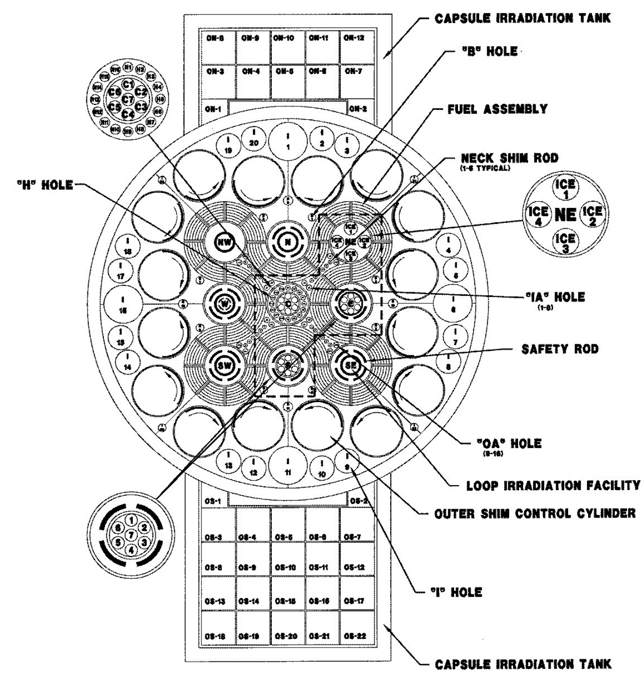

Figure 3.1: Advanced Test Reactor Core Cross-Section

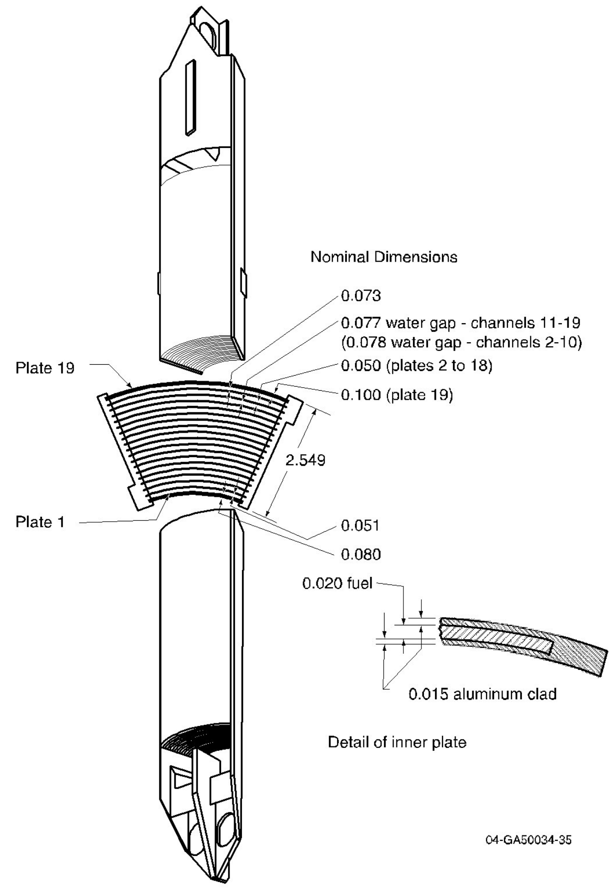

Figure 3.2: Advanced Test Reactor Fuel Element

3.4 ATR Fuel Design Basis

The Upgraded Final Safety Analysis Report (UFSAR) states that the reactor is designed so that its components meet the following performance and safety criteria [2]:

- The mechanical design of the reactor core components and their physical arrangement, together with corrective actions of the reactor control, and reactor protective systems (when applicable) ensure that the ATR Accident Analysis plant protection criteria are met.

- The fuel elements are designed to withstand various loads induced during handling and reactor operation

As stated in the UFSAR; the design basis for fuel element performance is as follows. The fuel element dimensions remain within operational tolerances during normal operation and the fuel element capabilities are not reduced below that assumed in the safety analysis. The extent of fuel plate failures remains within the limits assumed in the safety analysis. Fuel plate failure is defined as a breach in the fuel plate cladding allowing the release of fission products.

3.5 Previous ATR Hydraulic Tests

An extensive hydraulic development program was performed to investigate ATR fuel element performance and provide data to improve the design. Experiments and analytical work were as follows:

- Hydraulic buckling tests
  1. Vibration and Fatigue Test: During this test flow was incrementally increased in order to identify key harmonics associated with the fuel with provided flow rates [1-3].

  2. Hydraulic Buckling Test: flow was increased incrementally until measurable fuel element plastic deformation occurred [4, 5].

  3. Channel Blockage Test: an artificial channel blockage was simulated by adjusting a flow valve just prior to the inlet of the fuel element causing extreme axial pressure drops [2].

- Hydraulic proof tests
  1. Hydraulic testing of fuel elements fabricated in production lots was undertaken to assure that the fuel elements resulting from the fuel element research and development programs would be able to withstand the relatively severe operating conditions expected to be encountered in the ATR. [2, 6] These tests consisted of an examination of fuel plate and element deflection after being subjected to the most bounding limiting safety system settings (LSSS).

Furthermore, several experimental and analytical investigations were conducted in order to verify the hydrodynamic stability of the ATR under nominal and limiting conditions. As a result of these studies it was concluding that the ATR must operate under subcooled conditions in order to maintain stability due to potential burnout in the core [7-9].

3.6 Previous ATR Hydraulic Facilities

In 1959 construction of “The High Temperature Hydraulic Test Loop” began. The loop was a closed piping system designed for non-radioactive testing of production type ATR fuel elements. It was used initially to test the ATR production type fuel elements for adherence to the design requirements, to establish the acceptance criteria for production testing, and to study failure probabilities. The test loop design criteria are presented below in Table 3.1 [10].

After years of hydraulic proof testing the ATR fuel in the original hydraulic test to verify consistency in manufacturing capability the need for repeatability testing of fuel elements was removed from the ATR fuel fabrication qualification program. As a result of this the INL the original hydraulic test facility was deconstructed in 1994 in order to reduce on facility maintenance costs.

Table 3.1: Original Test Loop Design Criteria

Description	Value
Design Pressure [psig]	1100
Design Temperature Main Loop [F] Pressurizer [F]	 500 600
Operating Pressure Range [psig]	300 – 1000
Operating Temperature Range Main Loop [F] Pessurizer [F]	 150 – 500 70 – 550
Main Loop Pump Design Flow [gpm]	1200
Water Type pH Conductivity [micro-ohms] Oxygen Content [ppm]	 Demineralized 5.0 – 5.2 1 – 3 3.0

4 PROPOSED TEST LOOP

As part of the high performance research reactor fuel development hydraulic test loop the facility is required to produce a database of hydraulic information for all five HPRRs. Given that each high performance research reactor is unique, it is important to quantify all hydraulic characteristics of each reactor in order to determine what bounding properties the High Temperature Hydraulic Test Loop must have in order to satisfy all nominal and Limiting Safety System Setting (LSSS) conditions. It is also important to note that because this project is supporting the LEU conversion program, there is potential for geometric changes in fuel plate and fuel assemblies in several of the reactors. With this in mind all instrumentation and data acquisition equipment must be able to accommodate both current geometry and proposed LEU geometry.

4.7 Current HEU Hydraulic Characteristics

below identifies some of the critical characteristics of the HPRRs currently. For each characteristic presented in Table 4.1 the most bounding case (reactor) is highlighted. As a result of this information, while also considering potential geometry changes for the LEU fuels the design criteria for the High Temperature Hydraulic Test Loop can be established.

From Table 4.1 a single HPRR stands out using D_2O as its primary coolant while all others operate using H_2O. Although the NBSR operates under D_2O moderating and cooling conditions, it poses no significant problem for the potential test facility. The hydraulic characteristics of D_2O and H_2O (i.e. viscosity, density, thermal conductivity) remain within a relatively small difference under the operating conditions of the NBSR (i.e. ~7% or less).

Table 4.1: Current (HEU) HPRR Operating Conditions Summary

Parameter	MURR [11-14]	MITR [15-17]	ATR [2, 14]	HFIR [2, 14, 18]	NBSR [14, 19]
Reactor					
Thermal Power [MW_{th}]	10.0	5.0	250.0	85.0	20.0
Power Density [MW/m^3]	303	291	1000	1680	951
Enrichment [%]	93	93	93	93	93
Number of In Core Experimental Positions	6	3	68	37	4
Core					
Number of Fuel Elements	8	24	40	2	30
Number of Fuel Plate per Element	24	15	19	171, 369[i]	17
U-235 Content of An Assembly [kg]	0.775	0.506	1.075	2.60, 6.83[i]	0.350
Total Core Fresh Fuel Load [kg]	6.2	12.144	43	9.43	10.5
Fuel Assembly Geometry					
Overall Assembly Length [mm]	825.50	641.35	1684.276	635.0	1747.4413
Overall Fuel Plate Length [mm]	647.70	584.2	1257.3	609.6	330.2
Overall Active Fuel Length [mm]	609.60	558.8	1219.2	508.0	228.798
Overall Fuel Plate Width [mm]	43.26-110.287	64.16	51.55-99.97	81.0, 73.0[i]	70.942
Fuel Plate Thickness [mm]	1.270	2.032	1.270	1.270	1.27
Distance Between Fuel Plates [mm]	2.032	1.9812	1.9812	1.270	2.730
Fuel					
Material	UAl_x	UAl_x	UAl_x	U_3O_8	U_3O_8
Thickness	0.508	0.762	0.508	0.5334	0.508
Coolant					
Design Pressure [MPa][ii]	0.075	0.0	2.5642	3.227	0.0
Minimum LSSS Pressure [MPa][ii]	0.064	0.0	2.1553	2.5855	0.0
Bulk Coolant Design Temperature [°C] [iii]	48.9, 58.3	50, 55	51.6, 76.6	49, 85	37.8, 45.6
Bulk Coolant Maximum LSSS Temperature [°C]	136	60	93.3	140	48
Reactor Coolant	H_2O	H_2O	H_2O	H_2O	D_2O
Maximum Coolant Flow Rate Per Assembly [m^3/sec][iv]	0.2461	0.004995[v]	0.04598	0.8072[vi]	0.01893
Minimum LSSS Coolant Flow Rate Per Assembly [m^3/sec][iv]	0.01145	0.004732	0.00473	0.0122	0.01241

Bounding Hydraulic Parameter

4.8 Test Facility Summary

Based on the HPRR summary above (Table 4.1) a generalized "Required Design Envelop" of the Hydraulic Test Loop can be established for the key hydraulic parameters (loop pressure, coolant

i The two numbers represent the inner fuel assembly and outer fuel assembly respectively in the HFIR.
ii Pressures are representative of gage pressure
iii Values present nominal Inlet and exit coolant temperatures respectively.
iv Value represents volume flow rate through fuel elements.
v Value represents nominal flow rate, no maximum coolant flow rate specified.
vi HFIR Fuel Element is a single turbine like element, ~80% of the primary coolant passes through the element, therefore the total primary coolant flow in the reactor is 1.009 m^3/sec. The hydraulic testing will be conducted using 45° section of the total fuel element this produced a ~0.1009 m^3/sec maximum flow requirement through the test section

temperature, and coolant flow rate), these conditions are mapped out in Figure 4.1 and Figure 4.2:

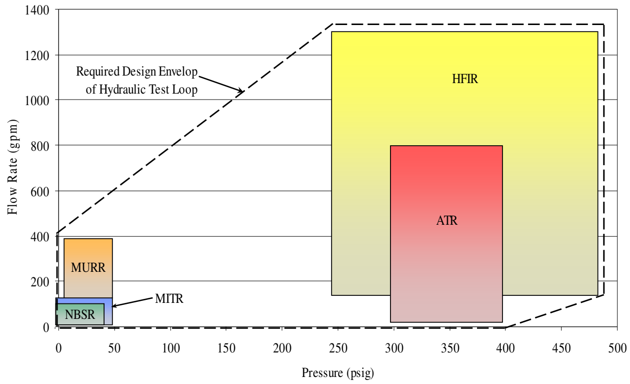

Figure 4.1: Reactor Design Pressure versus Flow Rate for all HPRRs

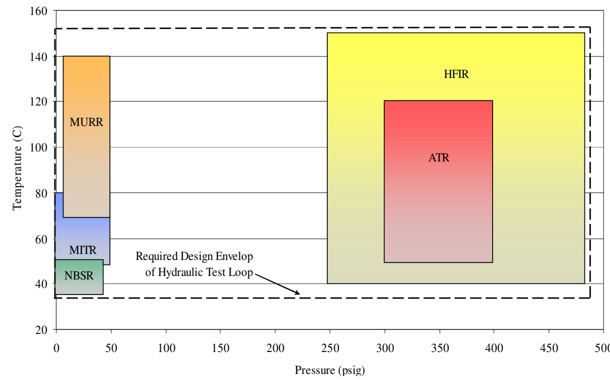

Figure 4.2: Reactor Design Pressure versus Temperature for all HPRRs

In order to acquire the design requirements that the hydraulic test loop must satisfy the most bounding hydraulic conditions must be identified from the figures above. As a result of this, the High Temperature Hydraulic Test Loop characteristics that satisfy these bounding conditions are outlined below in Table 4.2

Table 4.2: Design Loop Parameters

Description	Value
Design Pressure [MPa] (psi)	3.872 (561.6)
Design Temperature	
Main Loop [°C] (F)	200 (392)
Pressurizer [°C] (F)	300 (572)
Operating Pressure Range [MPa] (psi)	0.10133 to 3.227 (14.69 to 468.04)
Operating Temperature Range	
Main Loop [°C] (F)	45 to 180 (113 to 356)
Pressurizer [°C] (F)	45 to 280 (113 to 536)
Main Loop Pump Design Flow [m^3/sec] (gpm)	0.1009 (1600)
Water	
Type	Demineralized
pH	5.0 to 5.2
Conductivity	1 to 3 micro-ohms

A schematic of the conceptual test loop flow diagram is presented Figure 5.1. This flow diagram was developed through the evaluation of all individual research reactors' bounding conditions (seen in Table 4.1) as well as a preliminary study of the acquisition information required to be produced by the test loop in order to develop a sufficient database for fuel qualification purposes.

5 CONCLUSION

As a part of the RERTR fuel development program a hydraulic test loop will be designed, fabricated, and used to develop a database of hydraulic information which will be a part of the qualification of the newly proposed U-Mo LEU fuel for the HPRRs. This project is outline over a three year period and will conclude with the primary goal of implementing a single test facility to qualify hydraulic characteristics of the fuel for all HPRRs. By using a single facility the potential for both streamlining the safety analysis process during core conversions as well as improving the consistency in experimental data greatly increases.

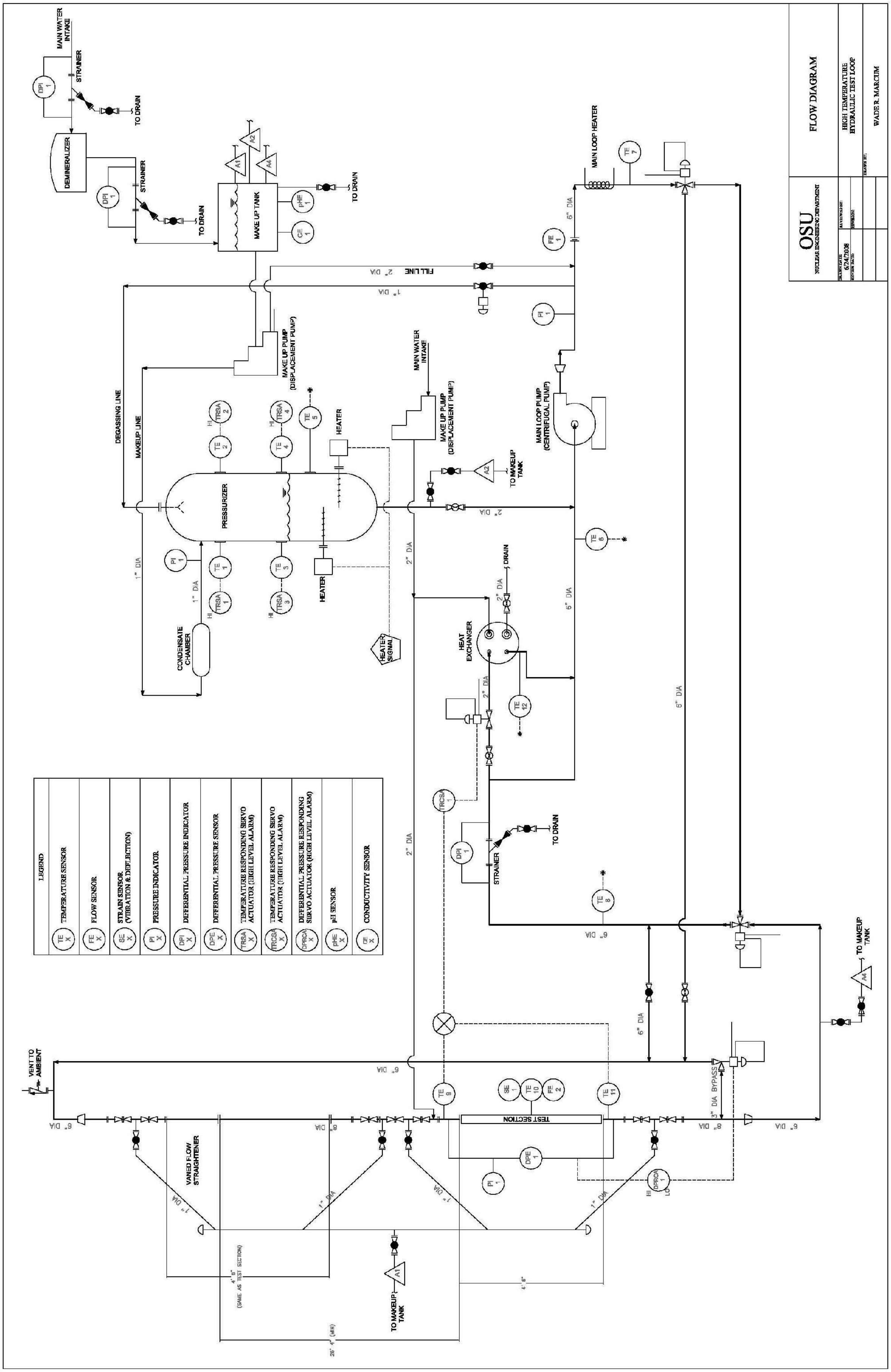

Figure 5.1: Preliminary Flow Diagram of Hydraulic Test Loop

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