Test Design Description for the Fusion Materials Open Test Assembly (Fusion MOTA-2A), Volume 1A

Part 1

R. E. Bauer

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TEST DESIGN DESCRIPTION
FOR THE FUSION MATERIALS OPEN TEST ASSEMBLY (FUSION MOTA-2A)
VOLUME 1A, PART 1

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TEST DESIGN DESCRIPTION FOR THE FUSION MATERIALS OPEN TEST ASSEMBLY (FUSION MOTA-2A), VOLUME 1A, PART 1

1.0 INTRODUCTION

This document encompasses the test requirements, hardware design, fabrication, and safety analysis for the Fusion Materials Open Test Assembly experiment for irradiation in FFTF Cycle 11 (Fusion MOTA-2A). Fusion MOTA is equally shared by the U.S. Fusion Material (DOE), Japanese Fusion Materials (MONBUSHO), and BEATRIX-II (IEA) programs. In the interest of providing optimum use of the irradiation space in the Fusion MOTA-2A and LMR MOTA-1G, eight of the Fusion MOTA canisters will be placed in MOTA-1G and an equal number of LMR canisters placed in Fusion MOTA-2A (Powell/Doran 1988). This eliminates the need to process Fusion MOTA-2A through the IEM cell prior to insertion for FFTF Cycle 11A.

The LMR MOTA design and safety analysis (Greenslade 1984) is the basis for much of this design and safety analysis report. This design description and safety analysis for the Fusion MOTA-2A is presented per the outline given in Chapter IV of the FTR User's Guide (Taylor 1978).

2.0 TEST PURPOSE AND OBJECTIVES

The purpose and objectives of the LMR program are discussed in the "Test Design Description for the Materials Open Test Assembly (MOTA): Design Description and Safety Analysis, Volume 1A" (Greenslade 1984).

2.1 U.S. FUSION MATERIALS PROGRAM

The fusion reactors of the future will expose materials to a unique environment which includes neutron exposure, elevated temperatures, and incident particle and heat fluxes. To ensure the safe, reliable, and cost effective operation of such reactors, the long term behavior of materials in the fusion environment must be understood. In some cases, materials will have to be developed to increase the attractiveness of fusion power.

The objective of the U.S. fusion materials program is to develop the technology base in materials for the successful implementation of fusion power in the next century. The objectives of the U.S. program in the FFTF are to investigate the impact of high neutron doses on material properties behavior, to develop low activation materials for fusion application, and to develop

fission/fusion correlations. The FFTF provides a unique facility for investigating materials behavior in a neutron environment because of its high neutron flux and its instrumentation capabilities which permit the irradiation of materials at controlled temperatures within the Materials Open Test Assembly.

2.2 MONBUSHO

The primary objective of the U.S.-Japan collaborative program is to enable the prediction of material behavior in magnetic fusion reactors based upon data from other irradiation facilities. Significant progress has been obtained in understanding material's response to fusion neutrons at low doses from the collaborative program at the RTNS-II facility. The collaborative effort now looks to the FFTF MOTA facility for extrapolating this understanding of materials behavior in a neutron environment to high neutron exposures.

The major goals of MONBUSHO's collaborative program in the FFTF/MOTA are:

- the development and verification of mechanistic models of damage production and evolution
- the ability to correlate the mechanical properties of materials irradiated in different environments based upon mechanistic models
- the ability to extrapolate low- or medium-fluence data to predict high fluence behavior.

The FFTF provides a unique facility for investigating materials behavior in a neutron environment because of its high neutron flux and its instrumentation capabilities which permit the irradiation of materials at controlled temperatures within the Materials Open Test Assembly.

2.3 BEATRIX-II

Fusion reactors require a supply of tritium to fuel operation. Since the world's tritium production is limited, it is necessary for these machines to generate their own supply. This is proposed to be accomplished by surrounding the plasma with a blanket containing lithium. The neutrons from the fusion plasma interact with Li to produce tritium and helium. The tritium is then recovered to refuel the reactor.

One form of blanket is solid breeders. These are made up of one of several types of lithium ceramics, a sweep gas to carry off the tritium as it is generated, cooling, and possibly a neutron multiplying material such as beryllium. It is important for blanket designers to characterize performance

of these ceramics under irradiation. Of particular interest are conductivity, tritium transport and release, and the structural integrity of the material.

The objectives of this test are specifically directed to the in situ evaluation of solid breeder materials performance with respect to multiple effects of irradiation damage, burnup, temperature and temperature gradient, and the local sweep gas environment. One performance characteristic of a solid breeder material which will be monitored is the in situ tritium release behavior. There has been some success in correlating time dependent tritium release predictions from diffusional theories with data taken from mixed spectrum in situ tritium recovery tests (Clemmer et al. 1984; Kurasawa et al. 1986; Verall/Miller 1987). The purpose of BEATRIX-II is to examine the kinetics of tritium release at higher burnup levels and more homogeneous burnup and heat generation conditions similar to those anticipated in ITER and commercial fusion blankets. The kinetics of tritium release are important because of the economics of fueling the plasma of a fusion reactor; tritium retained in the solid breeder may mean that tritium must be purchased from outside the fusion power plant.

The FFTF possesses sufficient test volume, neutron flux, and proven instrumentation capabilities for conducting BEATRIX-II. The neutron spectrum/flux associated with the FFTF provides a simulation of the nuclear heating and tritium generation rates anticipated in ITER and future fusion power reactors. Moreover, the ${}^6\text{Li}(n,\alpha){}^3\text{T}$ cross section in FFTF permits the testing of prototypic sized specimens with relevant temperature gradients and only modest self-shielding. Therefore, testing of more prototypic solid breeder material sizes can be conducted to relatively high lithium burnups.

The BEATRIX-II Cycle 11 experiment provides a demonstration for future tests in FFTF on other solid breeder materials, i.e., Canada's LiAlO_2 and the U.S.' Li_2ZrO_3 . The ability to monitor and control tritium containing gas flows will provide the opportunity for future testing which could include blanket module segment component testing. The homogeneous burnup and heat generation distributions in FFTF are well suited for this type of testing.

3.0 TEST AUTHORIZATION

3.1 U.S. FUSION MATERIALS PROGRAM

Approval-in-Principle for the Fusion MOTA was granted on April 6, 1988 (Hunter 1988).

3.2 MONBUSHO

The MONBUSHO program is authorized under Annex-I to the January 25, 1983 Exchange of Letters between the MONBUSHO of Japan and the Department of Energy of the United States of America on Cooperation in Fusion Research and Development. The title of Annex-I is "MONBUSHO - DOE Collaboration in Fundamental Studies of Irradiation Effects in Fusion Materials Utilizing Fission Reactors". The U.S. DOE will be the operating agency for this annex. Approval-in-Principle for the Fusion MOTA was granted on April 6, 1988.

3.3 BEATRIX-II

The BEATRIX-II experimental program is authorized under Annex-III to the IEA Implementing Agreement on a Program of Research and Development on Radiation Damage in Fusion Materials. The title of Annex-III is "Fast Flux Neutron Irradiation of Candidate Fusion Breeder Blanket Materials". The U.S. DOE will be the operating agency for this annex. Technical feasibility for the experiment was granted August 17, 1987 (Corrigan 1987). Approval-in-Principle for the Fusion MOTA was granted on April 6, 1988 (Hunter 1988).

4.0 TEST REQUIREMENTS

4.1 DESIGN

The specific design requirements for the U.S. and MONBUSHO portions of the test are included here and detailed in the "Engineering Test Plan for Fusion MOTA-2A" (Puigh 1988). The specific design requirements for the BEATRIX-II portion of the test are included here and detailed in the "Engineering Test Plan for BEATRIX-II, FFTF Cycle 11" (Bauer/Baker 1987). The design will be consistent with FFTF system interface requirements for open test assemblies. A general description of the MOTA and the requirements placed on the Fusion MOTA experiment are listed below.

4.1.1 Type

The Fusion MOTA will occupy instrumented open test position 2406.

4.1.2 Configuration

The MOTA, shown in Figure 1, consists of a Postirradiation Open Test Assembly (PIOTA) stalk attached to a control rod duct and a Multiple Test Assembly (MTA) inlet nozzle/orifice assembly. The control rod duct is attached

STRUCTURAL MATERIALS OPEN TEST ASSEMBLY (MOTA)

HEOL 0002-285.1

Figure 1. Materials Open Test Assembly (MOTA)

to the PIOTA stalk by a mechanical latch mechanism. Following irradiation, the latch is disengaged in the Interim Examination and Maintenance (IEM) cell and the PIOTA stalk can be raised out of the duct assembly to provide easy access to the test train.

The test train consists of a central hanger tube to which nine levels, each containing a set of up to six specimen canisters, are attached by hinged brackets. The first level of these canisters is located below the core region, the next five levels are located in the core region, and the final three levels are located above the core region. The canisters at the bottom of the test train are designated "below core" canisters; the in-core canisters occupy "Levels 1-5"; those at the top of the test train are designated "Levels 6-8". The Fusion MOTA test train is shown in Figure 2.

The MOTA is a standard irradiation vehicle which is described in the System Design Description (SDD 31 1975), Section 1.3.3. This Test Design Description covers the instrumented test subassembly and the interfaces with the MOTA. The analysis data covering the MOTA can be found in the following documents:

■ Specifications

- HWS-1506B, Part 3, Addendum 4
- HWS-1506B, Part 4, Addendum 4
- HWS-2655, MOTA Stalk & Test Duct Fabrication & Assembly Specification
- WHC-S-4007, Specification for 3/32" In-Line Check Valve for Materials Open Test Assembly (MOTA)

■ Drawings

- H-4-38176, Sheets 1-4, MOTA Experiment Interface
- H-4-33840, MOTA Stalk and Test Duct Assembly

■ Stress Reports

- SR-3, Addendum 4, PIOTA Modification B, Pressure Boundary Stress Report, August 1977
- SR-PIOTA-01, Addendum 4, PIOTA Modification B, Design Support Document, Preliminary Stress Analysis, August 1977
- ASME Stress Analysis for MOTA Quick-Connect Receptacle Assembly, SD-FI-ASME-00001 (Appendix B)

FUSION MOTA TEST TRAIN

MHC-SP-0224

38712-087.4

Figure 2. Fusion MOTA Test Train

■ Thermal/Hydraulic Reports

- ORE-089, WHC Document, "Radial Temperature Gradients in OTA Upper Load Pads During a Reactivity Insertion Transient", May 26, 1977
- ORE-137, WHC Document, "Radial Temperature Gradients in OTA Upper Load Pads During a Normal Scram", July 15, 1977
- ORE-105, "PIOTA Thermal Analysis Interim Report - Exit Ports", January 28, 1974, WHC Engineering
- ORE-167, "PIOTA Thermal Analysis Interim Report - Internal Shields", April 5, 1974, WHC Engineering
- ORE-168, "PIOTA Thermal Analysis Interim Report - Welded Insert Guide", April 5, 1974, WHC Engineering

■ Test Reports

- MOTA test reports as listed in ITMA letter 713, "MOTA/PIOTA Documentation", dated March 9, 1979

4.1.3 Performance

The test assembly will be compatible with reactor system operating conditions and will not be a safety hazard to the reactor.

4.1.4 Installation

The test assembly will be installed in the FFTF with the Closed Loop Ex-Vessel Machine (CLEM) using an approved detailed refueling plan. The Fusion MOTA electrical and gas line connectors will be connected in the center island using specially developed procedures.

4.1.5 Thermal/Hydraulic/Shielding

4.1.5.1 **Thermal.** The sodium outlet temperature of the test assembly will be controlled by the reactor sodium flow rate. The present MOTA flow rate of $14,000 \pm 2,000$ lbs/hr (Greenslade 1984) is adequate to meet the requirement for outlet temperatures established by the FTR User's Guide (refer to Section 7.0).

4.1.5.2 **Hydraulic.** The test assembly will be designed so the maximum pressure drop will be compatible with core pressure drops.

4.1.5.3 Shielding. Shielding is provided by the MOTA vehicle. No additional provision for shielding has been incorporated in the test train design.

4.1.6 Mechanical and Materials

The BEATRIX-II vented canisters will be mounted in a slotted hinge bracket so that they may be easily removed in the IEM cell after irradiation. This hinge bracket is similar to that used for the SP-100 capsules of MOTA-1F. Canister details are discussed in Section 6.0.

All Fusion MOTA vehicle and test train structural components that contact reactor sodium will be fabricated from Inconels and austenitic or ferritic stainless steels. Fusion MOTA specimens which are not compatible with sodium will be sealed in stainless steel capsules or other materials known to be compatible with sodium.

The below core canister of the standard MOTA will be replaced by six canisters similar to the standard MOTA canisters.

4.1.7 Design Life, Load Conditions, and Fluence Limits

All components in the test assembly will be capable of maintaining structural integrity during operation at full power and during design transients (as specified in the FTR User's Guide) over the projected design life of the tests. MOTA vehicle components will have a maximum design fluence limit of $1.7 \times 10^{23} \text{ n/cm}^2$ (Greenslade 1984). The lifetime of the MOTA, however, is currently limited by the differential swelling between the inner and outer walls of the canisters in the test train. Thus, maintaining target temperatures (not structural integrity) limits the lifetime of the MOTA.

4.1.8 Safety

The support and restraint of the test assembly and its contents will be designed to preclude reactor and assembly damage and to obtain the desired irradiation environment without affecting reactor operating conditions. The temperature control gas supply lines are designed to provide an adequate pressure boundary through the use of check valves which meet the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code.

The Tritium Handling System sweep gas lines between MOTA and the supply/analysis systems will be designed and fabricated to meet the requirements of the ASME Boiler and Pressure Vessel Code for Class 1 components. The reactor pressure boundary is terminated at Class 1 valves at the tubing entrance to the

Tritium Handling System glovebox (CM-10) and the tubing exit from the Sweep Gas Pressure Control Panel (CM-11).

4.1.9 Temperature Instrumentation and Control

The Fusion MOTA will utilize a temperature I&C system which is a duplicate of the existing MOTA system to measure and control canister temperatures. The Fusion MOTA I&C system will use the existing MOTA helium and argon control gas supply (Bauer 1987). The software to control Fusion MOTA will be based on the existing MOTA control software; however, modifications will be necessary for the BEATRIX-II canisters. The temperature control system is described in further detail in Section 6.0 of this document.

The control gas supply system will be designed and fabricated to minimize the impurities in the control gas.

The temperature I&C system is designed and fabricated to meet the following design criteria:

- Control capsule temperatures within $\pm 5^{\circ}\text{C}$ of set point
- Control known gas mixture to individual capsule (mix repeatable ± 0.1 percent, accurate ± 0.2 percent)
- Provide rapid helium purge capability of any or all capsules
- Provide manual control capability
- Provide flexibility to sense and change temperature limits and target temperature set points
- Provide out-of-limits annunciation
- Provide for gas valves to fail closed
- Provide temperature data history per capsule
- Provide capability to sample and record temperatures on all channels once/5 sec
- Provide history of mixture
- Minimize attended operation
- Maximize reliability through redundancy
- Maximize use of commercially available hardware

- Minimize interconnecting cables at FTR
- Measure and control temperatures for one MOTA (48 thermocouples, 31 valves)
- Interface with existing Type K thermocouples, blenders (primary and secondary), and gas distribution manifold
- Equipment should properly fit into compartment in such a manner as to facilitate ease of operations and maintenance, e.g., slide mounts and terminal strips should be utilized
- Compatible with environmental conditions on containment operating deck
- Automatic system is to consist of two identical systems connected in parallel to the thermocouples and valves (only one system active at a time)
- Automatic system to be capable of performing blends and gas distribution as directed by the operator (semiautomatic control)
- Operational capabilities should be provided for the following:
 - Individual canister blend calculations by the computer, at operators direction, in semiautomatic mode
 - Performing specific gas blends at specific intervals in automatic control mode
- Controller and backup controller to share communication link for control parameter updating
- Backup controller to be capable of real time temperature history plotting and hard copy for selected channels
- Backup controller to print out-of-limit conditions and create audible temperature alarms for channels above purge limit
- Controller to generate system alarms (audible) for inoperable equipment and is to include a display of current alarms
- Controller to measure and compare temperatures for 30 controlled thermocouples to prescribed limits once every 5 seconds
- Controller to measure and record temperatures for 18 remaining canister thermocouples and up to eight additional thermocouples once every 5 minutes

- Controller to record temperature for 30 controlled thermocouples at the following time intervals:
 - $T > T_{\text{purge}}$ - Every 5 seconds
 - $T_{\text{NH}} < T < T_{\text{purge}}$ - Every 60 seconds if delta-T from last reading $< 3^{\circ}\text{C}$
 Every 15 seconds if delta-T from last reading $\geq 3^{\circ}\text{C}$
 - $T < T_{\text{NL}}$ - Every 5 minutes if delta-T from last reading $< 3^{\circ}\text{C}$
 Every 15 seconds if delta-T from last reading $\geq 3^{\circ}\text{C}$
- Convert millivolt thermocouple readings to $^{\circ}\text{F}$ or $^{\circ}\text{C}$ as directed by operator; store readings as $^{\circ}\text{C}$
- Disk drive and storage format to be compatible with Hewlett Packard Model 9845, 85, and 200 series desk-top computers
- Controller to set blenders to ± 0.2 percent helium and ± 0.2 percent argon
- Manual control (full manual) solenoid valve operation to be accomplished by toggle switches (one switch for each valve) located on a panel on the mezzanine
- Valve position visual indication lights will be installed on mezzanine panel to indicate power was supplied to valves
- Primary and secondary solenoid valves are to be controlled from separate relay controllers
- Spare parts inventory to include at least one of each component of the system
- Additional controller located at 300 Area to be used for data reduction/analysis; sufficient software is to be developed to sort data from disk and store in condensed form; also, flexible plotting routines for report grade graphics
- Measure all BEATRIX-II temperatures within $\pm 2^{\circ}\text{C}$ accuracy
- Temperature of BEATRIX-II temperature change capsule to be maintained within $\pm 2^{\circ}\text{C}$ variation in a one hour period and $\pm 5^{\circ}\text{C}$ in a 24 hour period

- Control gas exchanges to BEATRIX-II temperature change capsule to be minimized to minimize disturbance of steady-state tritium permeation through subcapsule wall
- Temperature control system to provide input (analog or digital) of thermocouple readings and other system parameters to tritium sweep gas analysis I&C system
- Helium leak rate of gas supply to distribution manifold is to be less than 1×10^{-6} atm-cc/s
- Helium leak rate of individual gas lines from distribution manifold to reactor pressure boundary is to be less than 1×10^{-6} atm-cc/s

4.1.10 Interfaces

The interface requirements of the test assembly relative to other systems are given in Table 1.

4.1.11 Operability

The test assembly will be inserted in the reactor during the shutdown for Cycle 11A (S11A). It will remain in the reactor approximately 300 EFPD to meet goal exposures. All operating procedures will be established before Cycle 11A startup.

4.1.12 Maintainability

The test assembly is designed to remain in the reactor until goal exposures are reached. No interim examinations or reconstitutions are planned. The temperature I&C system is maintained by established procedures.

Maintenance and removal of waste from the BEATRIX-II tritium handling system will be by approved procedures to be in place before Cycle 11A startup.

4.1.13 Codes and Standards

Design, materials, fabrication, installation, and inspection of assembly hardware will conform to applicable standards and codes as given in the FTR User's Guide, Section V. All materials used for structural hardware in the MOTA will be procured to ASTM or NE standards. Materials used in nonstructural applications (e.g., canisters) will be procured to ASTM standards as specified in Volume 1B of the Test Design Description (Bauer 1988).

Table 1
FUSION MOTA INTERFACES WITH FFTF COMPONENTS AND SYSTEMS

<u>Item</u>	<u>Function</u>
Primary Sodium Coolant	Cooling for assembly and disposal of tritium released by permeation
Fuel Assemblies	Neutron and gamma environment
Shipping Casks	Shielded containers to ship irradiated specimens to hot cells for examination and/or testing
CLEM	Handling and transfer of MOTA
Fuel Failure Monitoring System	Means of identifying failed pressurized tube creep specimens and rupture times
Reactor Cover Gas	Disposal of purged helium, argon, and tritium released by permeation
IEM Cell	Facility for reconstituting specimens into MOTA prior to irradiation and specimen/canister removal after irradiation
MOTA Vehicle	Vehicle for in-reactor irradiation of test train experiments
CAPS	Tritium handling system effluent disposal
Shipping Containers	Shipment of tritium handling system tritium collection media for disposal
FTR Control Room	Alarms
FTR Air Monitors	Tritium alarms must be in place and operable prior to Cycle 11A startup
Two Thermocouples	Routed to plant data system to measure outlet bulk sodium temperature

The ASME codes will specify the allowable stresses for the test assemblies. Where possible, the Nuclear Systems Materials Handbook (NSMH) will be the source of material properties data used in design and subsequent analyses. Reactor pressure boundary portions of the MOTA will be designed and fabricated to the requirements of the ASME Boiler and Pressure Vessel Code, Section III.

4.1.14 Surveillance and In-Service Inspection

Subcapsule temperatures will be controlled from an experimenter control console and will be fully automatic with manual backup control. A failure of the console will activate an alarm. Procedures for console failure situations will be similar to those currently in place. Procedures to accommodate potential capsule/thermocouple failures, as well as tritium handling system alarms, will be completed prior to Cycle 11A startup.

4.1.15 Quality Assurance

Quality assurance for this irradiation experiment will be in accordance with the requirements of ANSI-ASME NQA-1. Details which describe the Quality Assurance program are included in Volume 1B (Bauer 1988) of this Test Design Description as well as the "Engineering Test Plan for the Fusion MOTA" (Puigh 1988) and the "Engineering Test Plan for BEATRIX-II, FFTF Cycle 11", (Bauer/Baker 1987).

4.1.16 Tritium Data Acquisition System

An instrumentation system will also be installed for the BEATRIX-II experiment. The purpose of the system will be to monitor and record the data from the BEATRIX-II vented capsules and to generate system alarms. The thermocouples from the BEATRIX-II canisters will be connected in parallel between the temperature control and BEATRIX-II systems.

Requirements for the tritium data acquisition system are as follows:

- Instrumentation will be provided to measure the parameters listed in Table 3 within the specified accuracy and frequency
- Transducers and sensors which contact the sweep gas containing tritium will be located within a glove box
- Output voltages from the instrumentation will be compatible with the input requirements of the dedicated data acquisition system
- Essential components of the data acquisition system will be installed in duplicate
- Data acquisition system will be capable of displaying output from a single channel in real time
- Data acquisition system will have a minimum of 100 input channels for Cycle 11 with capability for expansion to 200 channels

- Scan rate will be at least 10 channels per second
- Output data will be recorded on magnetic storage media
- An off-line computer will be provided for data analysis, software development, and data conversion to other formats
- Strip chart recorder(s) will be provided to simultaneously record two channels of input data
- Instruments are to be compatible with radio frequencies encountered in FFTF containment

4.1.17 Tritium Handling System

- Provide individually controlled sweep gas to the two monitored capsules for flows ranging from 10 to 1000 cc/min and controlled within 2 percent of set point
- Sweep gas for the ring specimen and solid pellet is to be helium which can be doped with 100, 1000, 3000, or 10,000, ppm hydrogen; gas will be certified to be within 10 percent of desired composition
- Tritium in the recovery system effluent will average <6.5 mCi/day
- All tritium containing lines are to be joined by welding, Cajon VCR fittings, or copper gasketed metal flanges such as the Varian conflat flange
- The sum of impurities in the sweep gas inlet will be less than 10 vppm excluding hydrogen with typical impurity limits as listed in Table 2

Table 2
SWEEP GAS IMPURITIES LIMITS

<u>Element</u>	<u>Limit (vppm)</u>
N ₂	3
O ₂	1
H ₂ O	1
Ar, Ne, others (Excluding H ₂)	5

- Instrumentation requirements for the tritium handling system are shown in Table 3
- Ion chambers will have the minimum volume consistent with the requirements of Table 3, be calibrated with tritium gas, and pressure controlled within 5 percent
- Design of the tritium handling system will be in accordance with ALARA goals
- Helium leak rate on nontritium containing gas lines is to be less than 1×10^{-8} atm-cc/s

4.1.18 BEATRIX-II Experiment General Requirements

- Space for two standard size MOTA capsules for tritium monitored capsules in Level 1
- Space for eight standard size MOTA capsules for nonvented subcapsules; nonvented subcapsule test matrix will be detailed in Volume II of this Test Design Description
- Twenty-six of the seventy-eight penetrations through the IPS bulkhead will be available for BEATRIX-II
- Temperature control gas lines for one tritium monitored capsule
- Helium sweep gas lines for both monitored gas lines (100 cc/min helium at STP)
- Estimated combined tritium production for all BEATRIX-II specimens is not to exceed 100 Ci/day for Cycle 11

4.1.19 BEATRIX-II Nonvented Subcapsules

- Subcapsules will be fabricated from materials and geometries within existing WHC qualified weld procedures where practical
- Nonvented subcapsules will be included in standard MOTA capsules at positions 1B, 1D, 5E, and 8B

Table 3
TRITIUM HANDLING SYSTEM INSTRUMENTATION REQUIREMENTS

<u>Parameter</u>	<u>Range</u>	<u>Accuracy</u>	<u>Frequency</u>
Total tritium concentration in sweep gas	0-10 Ci/m ³ 0-100 0-1000 0-10000	±10% F.S.	10 seconds
HT tritium concentration in sweep gas	0-10 Ci/m ³ 0-100 0-1000 0-10000	±10% F.S.	10 seconds
Sweep gas flow rate	0-500 cc/min	±1% F.S.	10 seconds
Sweep gas inlet pressure	0-100 psig	±4 psi	10 seconds
Sweep gas outlet pressure	0-100 psig	±4 psi	10 seconds
Sweep gas inlet O ₂ concentration	0-100 ppm	±1 ppm	10 seconds
Sweep gas inlet H ₂ O concentration	0-100 ppm	±1 ppm	Every bottle change
Sweep gas outlet H ₂ O concentration	0-100 ppm	±1 ppm	10 seconds
Recovery system effluent tritium concentration	0-0.001 Ci/m ³ 0-0.01 0-0.1 0-1	±10% F.S.	10 seconds
Glove box atmosphere tritium concentration	0-0.001 Ci/m ³ 0-0.01 0-0.1 0-1	±10% F.S.	10 seconds
Glove box pressure	-5 - +5* in H ₂ O	±0.5 H ₂ O	10 seconds
Total tritium concentration outside of glove box	0-10 uCi/m ³ 0-100 0-1000	±10% F.S.	10 seconds

*Relative to containment pressure.

- The solid breeder material will be loaded and handled in an inert atmosphere (<2 ppm H_2O) at all times after final outgassing
- Final outgassing will be performed in $25^\circ C$ temperature steps to $600^\circ C$; the pressure shall be allowed to stabilize at 1×10^{-5} torr or less between subsequent steps
- Subcapsules will be sized such that internal pressures will not cause the subcapsule to breach

4.1.20 BEATRIX-II Temperature Change Capsule

- Capsule will be located in position 1-E
- Breeder material to be isolated from reactor sodium coolant
- Subcapsule and breeder materials to be compatible
- Temperature is to be measured internally at two points and externally at one point
- Sweep gas lines and subcapsule are to be designed for a calculated tritium permeation loss of less than 1 percent of the anticipated steady-state release rate
- The minimum temperature of the inner surface of the temperature change sample is $450^\circ C$ or less
- The maximum temperature of the inner surface of the temperature change sample is $650^\circ C$ or greater
- End-of-life total lithium burnup of 4 percent in Li_2O is desired
- Breeder material is to be Li_2O enriched with 6Li to achieve a balance of the burnup and temperature requirements
- Radial temperature differential will be less than $75^\circ C$ at the beginning of operation
- The predicted temperature difference between the specimen midplane and the top or bottom will be less than $25^\circ C$
- The solid breeder material will be loaded and handled in an inert atmosphere (<2 ppm H_2O) after final outgassing.

- Final outgassing will be performed in 25°C temperature steps to 600°C; the pressure will be allowed to stabilize at 1×10^{-5} torr or less between subsequent steps
- The relative local flux level near the canister will be measured by a self-powered neutron detector (SPND) of similar device with a sensitivity of ± 3 percent
- Dosimetry for postirradiation determination of the cycle average neutron spectrum will be placed within 4 cm of the specimen radially and will cover the entire axial length of the specimen

4.1.21 BEATRIX-II Solid Pellet Capsule

- Capsule will be located in position 1-C
- Breeder material is to be Li_2O enriched with ^6Li to achieve a balance of the desired burnup and temperature
- The goal end-of-life total lithium burnup is 4 percent in Li_2O
- Breeder material to be isolated from reactor sodium coolant
- The design target pellet centerline temperature to be 900°C at beginning of operation
- Pellet outer surface temperature to be 450°C or less at beginning of operation
- The predicted centerline temperature at the top and bottom of the specimen column is to be greater than 600°C; the temperature profile is to be as flat as possible within practical limits
- Subcapsule and breeder materials to be compatible
- Temperature to be measured internally at two points and externally at one point
- Sweep gas lines and subcapsule are to be designed for a calculated tritium permeation loss of less than 1 percent of the anticipated steady-state release rate
- The temperature measurement capability of the interior thermocouples will be at least 1100°C
- The solid breeder material will be loaded and handled in an inert atmosphere (< 2 ppm H_2O) at all times after final outgassing

- Final outgassing will be performed in 25°C temperature steps to 600°C; the pressure will be allowed to stabilize at 1×10^{-5} torr or less between subsequent steps

4.2 TEST ENVIRONMENT

The MOTA will be located in Row 4 OTA, position 2406. The assembly is instrumented and the design and specimen placement is based on neutron flux (Nelson 1987) and nuclear heating rate calculations for these positions at a reactor power level of 294 MW for FFTF Cycle 9A and scaled for Cycle 11A position 2406 (Bauer 1988b). Goal fluences for a 300 day operating cycle range between 6×10^{20} n/cm² (Level 8) and 7.0×10^{22} n/cm² (Level 3) ($E > 0.1$ MeV). The design peak nuclear heating rate in stainless steel is 4.0 W/g (Bauer 1988b). Figure 3 shows the location of the Fusion MOTA within the FFTF core as well as the radial designations for the canisters. The proposed canister program and temperature assignments are shown in Table 4. The actual canister assignments will be included in Volume II of this Test Design Description.

4.3 HANDLING AND SHIPPING

The Fusion MOTA will be shipped from the 306E Building in the 300 Area to the FFTF in a WHC-supplied shipping container. The standard shipping procedure for MOTA is adequate for this experiment. Assembly identification will conform to the requirements set by the FTR User's Guide. Permission to ship will be obtained from the FFTF Irradiation Test Management (ITM). Any specific handling procedures and receiving inspection requirements required at that time will be supplied prior to the shipment of the test assembly.

4.4 ASSEMBLY AND DISASSEMBLY

Hardware fabrication and assembly of the Fusion MOTA will be performed in the 306E Building by WHC Test Engineering and Development in accordance with Volume 1B of this Test Design Description (Bauer 1988a). Final specimen loading and experiment assembly will be performed prior to shipment of the Fusion MOTA to FFTF.

Postirradiation removal of the experiment canisters will be performed in the IEM cell in accordance with an approved work plan as specified in Volume 1B of the Test Design Description. Casks normally used for transporting irradiated materials will be adequate for the transfer of Fusion MOTA canisters from FFTF to the hot cell facilities in the 300 Area. Removal of the BEATRIX-II vented canisters from the MOTA stalk in the IEM cell will require special tooling and procedures to sever and seal the sweep gas lines. The lines are to be sealed to prevent moisture and other contaminants from reaching the breeder material specimens.

FFTF CORE MAP

WHC-SP-0224

38712-087.3

Figure 3. Fusion MOTA Te Article Orientation

Table 4
FUSION MOTA-1/LMR MOTA-1G CANISTER ASSIGNMENTS

MOTA-2A (FUSION MOTA) CANISTER MATRIX

	A	B	C	D	E	F
8	440°C FUSION [offset] W (HHF)	440°C FUS/MONB/ IEA W				
7		435°C FUS/MONB W				
6	430°C FUSION [offset] W (HHF)	430°C FUS/MONB W	430°C MONB [offset] W (HHF)			
5	600°C FUSION [offset] W (HHF)	425°C FUSION [offset] W (HHF)	600°C FUSION [offset] W (HHF)	800°C MONB	425°C IEA/MONB W	600°C MONB
4	425°C LMR W	600°C FUSION	600°C FUSION	425°C MONB	520°C MONB	600°C MONB
3	605°C LMR	550°C LMR	420°C FUSION W	420°C MONB W	520°C MONB	600°C MONB
2	410°C LMR W	410°C LMR W	520°C FUSION	410°C MONB W	410°C MONB W	520°C MONB
1	390°C MONB W	390°C IEA W	900°C IEA [3 TCs] W	390°C IEA W	450-600°C IEA [3 TCs] W	460°C MONB
BC	365°C FUSION W	365°C FUSION W	365°C LMR W	365°C LMR W	365°C MONB W	410°C MONB

W = Weeper Canisters
FUSION = U.S. Fusion Program
MONB = Japanese MONBUSHO Program
IEA = I.E.A. BEATRIX-II Experiment
LMR = Liquid Metal Reactor Program
HHF = High Heat Flux Capsule
[offset] = Offset Thermocouple

MOTA-1G CANISTER MATRIX

	A	B	C	D	E	F
8					440°C FUS/LMR W	440°C LMR [offset] W
7						
6					490°C LMR	435°C LMR [offset] W
5	670°C LMR [offset]	550°C LMR	550°C LMR	635°C LMR	670°C LMR [offset]	430°C PNC W
4	600°C FUSION	635-605°C LMR	750°C LMR	550°C LMR	670°C LMR	670°C LMR
3	420°C FUSION W	420°C FUSION W	605°C PNC	670°C PNC	550°C LMR	750°C LMR
2	520°C FUSION	605°C LMR	635°C LMR	550°C PNC	410°C LMR W	520°C FUSION
1	750°C LMR [offset]	495°C LMR	495°C LMR	495°C PNC	390°C LMR W	390°C LMR W
BC	365°C LMR [no TC] W	365°C FUSION [no TC] W	365°C LMR [no TC] W	365°C LMR [no TC] W	365°C LMR [no TC] W	365°C FUSION [no TC] W

W = Weeper Canisters
LMR = Liquid Metal Reactor Program
PNC = Japanese PNC Program
FUSION = U.S. Fusion Program
[offset] = Offset Thermocouple

Nominal
Mid-Canister
Fast Fluence
(10²² n/cm²)

0.045

0.17

0.54

3.5

6.3

7.5

6.8

3.9

1.2

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4.5 OPERATION AND MAINTENANCE

Overall operation and maintenance of the Fusion MOTA will be the responsibility of FFTF Operations. Test requirements and procedures for insertion, irradiation, and transfer of the experiment to the IEM cell will be referenced in Volume II, Part 1, of the Test Design Description which will be submitted after fabrication and assembly of the Fusion MOTA is completed.

Operation of the tritium handling system will be the responsibility of FFTF Operations, under the direction of Irradiation Development. Operation of the tritium handling system will be per approved procedures and test plan.

Maintenance of the tritium handling system is the responsibility of FFTF Operations and will be performed per approved procedures.

4.6 STANDARD PARTS AND MATERIALS

The experiment will be assembled into a standard MOTA vehicle. The MOTA vehicle is comprised of several standard components, some of which are modified. Table 5 lists the MOTA vehicle standard components.

Table 5
STANDARD PARTS REQUIRED FOR ASSEMBLY OF FUSION MOTA

<u>Drawing No.</u>	<u>Item</u>	<u>Quantity</u>	<u>Item</u>
H-4-33958	1	1	Piston Ring
H-4-21718	1	1	Duct Tube, Control Rod
H-4-29958	1	1	PIOTA Assembly
H-4-66182	1	1	MTA Nozzle Body
H-4-66188	1	1	MTA Shield Housing
H-4-66190	1	1	MTA Shield Plug Casting

4.7 EXAMINATION

No interim IEM cell examinations are required for Fusion MOTA.

4.8 PLANS AND PROCEDURES

Plans and procedures describing pre- and postirradiation handling and assembly/disassembly of the Fusion MOTA will be provided by Irradiation Development for inclusion in the Detailed Irradiation Plan.

melting the residue sodium in the annular space between end cap and capsule tube. The results of this test are reported in "Sodium Testing of Breeder Reactor Structural Materials Experiment Specimen Capsule" (Maffei 1978).

- Hydraulic Water Flow Test - Several tests were performed using a water flow loop to measure the pressure gradients along the sodium channels among the capsules. Pressure taps located in the hex tube duct section allowed the precise measurement of pressure gradients. The total pressure drop across the core section was measured at less than one psia at a flow rate of 14,000 lb/hr equivalent sodium flow (30 gpm). The flow loop used a full scale test section mockup to provide all the necessary and important wetted surfaces. View windows in the hex duct tube allowed visual examination of flow characteristics and capsule filling/draining phenomena under dynamic conditions. Plexiglass capsules were utilized to design optimum capsule configurations. Results of dynamic testing showed that the capsules will refill in less than 7 seconds after complete voiding. The refill time was reduced from the static test results by the dynamic pressure gradient. This testing is discussed in Appendix I.
- Gas and Thermocouple Lead Mechanical Testing - Two tests were conducted on the gas and thermocouple leads:
 - (1) A test was performed in the laboratory to ensure that the gas lines and thermocouple located at the base of the capsule could be flexed when the capsule was rotated about the hinge pin. This test showed there was very little inelastic bending of these leads and that they were capable of withstanding many cycles. The number of cycles during the life of the assembly will be less than 10.
 - (2) A lead compression test was performed to ensure that the loads transmitted to the pressure boundary are within the allowable loads specified for the MOTA. The test is reported in "Lead Compression Test for the Breeder Reactor Structural Material Experiment" (Krogness 1978).
- Sodium Loop Test - The feasibility of the final design was verified with a full scale MOTA sodium loop test. This test was instrumented and performed similarly to the water flow test previously described in this document. Instrumentation was provided to measure sodium pressure and temperature. Reactor gamma heating was simulated with internally mounted electric heaters. Specimen ruptures were simulated by injecting small quantities of helium gas into the capsule. Results from the first phase of testing were used to make a design change to improve the capsule performance. This design improvement was accomplished by adding a formed seal to the

4.9 DISPOSITION

The Fusion MOTA hardware will be used for one 300 EFPD (@ 294 MW) cycle. After goal exposures are reached, it will be discarded and replaced with new hardware. Specimen canisters will be forwarded to other hot cell facilities as described in Section 4.4 of this document.

5.0 TEST PROGRAMS AND SCHEDULES

5.1 DEVELOPMENT PROGRAMS

The complexity of the MOTA required the performance of several tests to verify design feasibility and operability:

- Gas Fill and Exchange Test - Tests were performed to determine the amount of gas and time required to exchange a mixture in the capsule annular gap. This was accomplished by constructing a prototypic double-wall capsule with an electrical heater and thermocouples to measure internal temperatures. One gas mixture was replaced with another in less than 5 seconds and three volumetric exchanges. This same test apparatus was connected to an automatic controller and gas blender to simulate one possible control system configuration. Manual adjustment of heater power resulted in a capsule temperature change and a response by the controller to supply a new gas blend and return the temperature to the desired limit. Various control algorithms were tried and all were successful. The system proved to be very stable.
- Sodium Drain Test - A sodium drainage test was conducted to determine the relationship between an orifice size and gravity drain time of sodium. This information was used to specify a minimum hole diameter for the sodium inlet and outlet tubes so reasonable drain/fill times could be obtained without the possibility of blockage. A prototypic capsule was constructed and tested in sodium to determine static fill and drain times and the difficulty in removing the top end cap after it was "wetted" with sodium. The results of this test showed that a capsule will drain in less than 16 seconds with very little residue sodium left in the capsule. Likewise upon being resubmerged, the capsule rapidly refilled in approximately 23 seconds. It was discovered during the test that the drain line did not completely drain. The small amount of residue sodium froze and formed a plug at room temperature. An additional 15 seconds at 400°F was required to melt the sodium plug in the drain line. The testing verified that the end cap can be easily removed with the application of heat by a hot "air" gun

removable cap to eliminate excessive annular sodium flow between the cap and the capsule. Subsequent testing successfully demonstrated the adequacy of the design. The instrumented capsules consistently vented and refilled with sodium in a simulated specimen rupture within 20 seconds. Temperatures could be varied by a factor of four in the gas gapped electrically heated capsules.

At the completion of the sodium loop testing, the test vehicle and test train were used to demonstrate the removal process from the hex tube. It was necessary to heat the hex tube and melt the residue sodium before removing the test train. Capsule latch hinges operated easily for specimen removal. A full report of this test has been issued, "Breeder Structural Materials Experiment/Materials Open Test Assembly (BSME/MOTA) Sodium Loop Test Report" (Marquardt 1979).

- Gas Blend/Control System - The control system was assembled by WHC and has undergone an acceptance test phase. Included in the acceptance testing was the connection of the system to a prototype which simulated the in-reactor response of the MOTA. The control system was tested to ensure that the proper gas mixtures are specified and delivered when required.
- Vacuum Insulated Endcap Water Flow Test - Full scale water flow tests of the vacuum insulated endcap were conducted to determine the leakage flow through the capsule specimen region. The testing was conducted for caps with 0.069" and 0.054" inside diameter vent tubes. The results of the testing are used in designing the "shim" at the bottom of the capsule to preheat the incoming sodium to the desired temperature. This testing is discussed in detail in Appendix G.

5.1.1 U.S. Fusion Materials Program

Canisters for the U.S. Materials program do not require additional development testing.

5.1.2 MONBUSHO

Canisters for the MONBUSHO portion of Fusion MOTA do not require additional development testing.

5.1.3 BEATRIX-II

The BEATRIX-II experiment will use two specially developed canisters. These are described in detail in Section 6.0 of this document. The ring specimen temperature control canister requires no special development testing. The solid specimen canister requires development testing for both fabrication of the channeled nickel layer on the capsule wall inside diameter as well as testing to characterize the gas flow through the channeled layer. This testing is discussed in Appendix K.

5.2 ANALYTICAL PROGRAMS

Several types of analyses were conducted to provide technical design support of the original MOTA:

- Orifice sizing
- Gas line check valves
- Hydraulic flow (test duct assembly)
- Thermal
- Stress
- Safety

The results of these analyses are presented in Sections 7.0 and 8.0 of this document.

Several analyses were conducted to provide technical design support for Fusion MOTA:

- Canister thermal analysis
 - Design
 - Independent verification
 - Thermal hydraulic analysis
 - Full MOTA
 - Individual canisters

- Stress analysis
 - BEATRIX-II canisters (new designs)
 - Safety analysis

The results of these analyses are presented in Sections 7.0 and 8.0 of this document.

5.3 OVERALL TEST SCHEDULE

The test schedule for Fusion MOTA is shown in Table 6.

Table 6
TENTATIVE FUSION MOTA SCHEDULE

Conceptual Design Review	November 1987
Design Review (BEATRIX-II Working Group)	March 1988
Final Design Review	May 1988
Submit TDD-1A/1B	May 1988
Delivery of Gas Analysis/Recovery System	December 1988
Installation of Gas System	January 1989
Checkout of Gas System and Operation Review	February 1989
Complete MOTA Assembly Fabrication	January 1989
Start Irradiation	April 1989
Remove Test from Reactor	June 1990
Remove Canisters from MOTA Stalk	July 1990

6.0 TEST DESCRIPTION

6.1 CONFIGURATION AND DESIGN PARAMETERS

The Fusion MOTA vehicle is similar to the LMR MOTA vehicle which has been irradiated in FFTF Cycles 2-10. The differences between the two are in the test train. The major differences in the Fusion MOTA are:

- "Standard" type canisters in the below core position
- Extension of the support stalk through the below core position
- Sweep gas lines which will flow gas first into a specimen canister and then out of the reactor vessel to an analysis system

- Relocation of ASME check valve in temperature control gas lines
- Modified instrumentation connector

The experimental test train consists of multiple specimen capsules clustered on nine elevations in groups of six. Figure 1 shows the arrangement of the standard reactor components with the experiment housed in the test duct. Each cylindrical capsule in Levels 1-8 is approximately 7.5 inches long by 1.326 inches in diameter. The below core capsules are approximately 6.5 inches long by 1.326 inches in diameter. Figure 2 shows the cluster arrangements in more detail. All of the capsules are attached to a central support tube by a simple pivot hinge. A small locking latch retains the capsules in a closed position. Access for typical specimen loading and removal is provided by the pivot action of each capsule after the hex duct is removed from the PIOTA stalk. Instrumentation is routed within the support tube which is attached to the PIOTA stalk.

Specimen test temperatures are measured with a metal sheathed, MgO insulated, chromel-alumel thermocouple. The junction of each thermocouple is located at the midplane of the canister test volume. Typically the thermocouples exit the lower end of each canister, bend around the pivot hinge to permit flexing, and are then routed up through the support tube and PIOTA stalk. Within the PIOTA stalk, the thermocouples are coiled to allow for thermal expansion. The thermocouples must penetrate the reactor pressure boundary and then terminate at the upper end of the PIOTA. Each thermocouple is attached to the electrical connector by flexible sheathed leads.

Standard capsules are of two designs. The simplest is a weeper and is shown in Figure 4. It allows reactor coolant sodium to enter the specimen compartment and flow around and among the test specimens. Temperatures are measured with one centrally located thermocouple. Actual test temperatures are dependent on the local sodium coolant temperature.

The second standard capsule type is more complex and is shown in Figure 5. It provides a means to elevate test temperatures above ambient sodium temperature as well as a method to maintain and control temperatures during reactor operation. These capsules are of double-wall construction, similar to a Dewar flask. Utilizing nuclear heating in the specimen, surrounding sodium, and capsule structural components as heat sources and by properly sizing the insulating gas gap, a thermal balance is achieved at a specified operating temperature. A simple energy balance was used to determine the appropriate gap dimensions. A one-dimensional energy balance for this system is:

$$Q_{in} = Q_{out}$$

where: Q_{in} is the energy generated within the boundary of the gas gap.

Q_{out} is the energy transported across the gas gap by conduction and radiation.

Applying the heat generation, conductivity, and radiation terms, the fully developed one-dimensional steady-state equations per unit length become:

$$\frac{2\pi K (T_3 - T_4)}{\ln R_4/R_3} + \frac{\sigma A(T_3^4 - T_4^4)}{1/\epsilon_1 + 1/\epsilon_2 - 1} - m\gamma = 0$$

Conduction

Radiation

Heat
Generation

- where: m = Mass subject to energy generation per unit length.
- γ = Local nuclear heating per unit mass of material.
- K = Average thermal conductivity per unit length of the annular gap.
- T_3 = Temperature of the inner side of the annular gap.
- T_4 = Sink temperature of outer side of the annular gap.
- R_4 = Outside radius of the annular gap.
- R_3 = Inside radius of the annular gap.
- σ = Stefan-Boltzmann constant for radiation heat transfer.
- A = Inner surface area of the annular gap per unit length.
- ϵ_1, ϵ_2 = Emissivity coefficients for the capsule walls on each side of the gap.

Typically, gaps run between 0.008 and 0.045 inch depending on the local gamma heating rate, coolant temperature, and specimen loading density.

Temperatures in each gas gapped capsule can be adjusted by changing the conductivity of the insulating gas. The gap will contain a mixture of argon and helium. There is approximately a factor of seven difference in thermal conductivities of helium and argon; therefore, theoretically a factor of seven change in power at constant temperature or a factor of seven change in gas gap temperature difference at constant power should be possible. Due to a complex

(BELOW CORE USEABLE SPECIMEN LENGTH = 5.08 INCH)

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Figure 4. Fusion MOTA Weeper Canister

38712-087.2

Figure 5. Fusion MOTA Gas Gapped Capsule

combination of radiation, convection, and axial heat losses as well as gap closure due to thermal expansion, this control factor can be as low as four.

Exchanging the gas in the annulus is made possible by routing a 3/32 inch OD capillary tube through the PIOTA to the bottom plenum of the capsule. The in-flowing gas pushes the "old" gas out through an exit vent tube at the capsule top. The exit line is routed down the outside capsule wall, around the pivot hinge, and back up the support tube to the PIOTA stalk. The vent tube terminates in the PIOTA stalk above the reactor sodium pool. The purge gas mixture is vented directly to the reactor sodium cover gas. Each capillary tube for temperature control penetrates the reactor pressure boundary through an ASME coded check valve. Specially designed electrical and gas line connectors mounted on top of the PIOTA join all the gas and electrical instrumentation. The ASME coded check valves are included as part of the quick disconnects located in the gas line connector. The check valve is made by modifying a standard Swagelok quick connect such that the poppet valve operates as a check valve.

The double-walled capsules are designed to be self-filling/draining with reactor coolant sodium. Sodium enters through a 0.147 inch inner diameter (ID) tube into the specimen chamber. Gas is vented through a vent line located in the removable top end cap which can be either 0.054 or 0.069 inch ID. The end cap is evacuated to minimize axial heat loss. The hydraulics of the capsule allow rapid sodium filling during reactor insertion. Likewise, in the event of a pressurized specimen rupture, the released gas will be voided rapidly to minimize capsule overheating. In addition, the inlet and outlet orifices were sized to be smaller than the smallest debris (a specimen end cap) which would be generated from any rupture.

Once the capsule has filled, the transport sodium flow through the capsule is limited by the high hydraulic resistance in the end cap. The flow characteristics of the end cap are discussed in Appendix G.

A two-dimensional computer code analysis was performed and the annular gaps near the capsule ends were tailored to minimize thermal gradients. The analysis was based upon a homogeneous specimen mass loading and uniform gamma heating in the radial and axial directions. Between the tailored capsule ends, the annular gap is "sized" by a one-dimensional thermal analysis based upon inputs from the experimenter. This analysis is performed just prior to final assembly which permits the greatest flexibility to the experimenter in selecting capsule loading configuration. The annular gap is "sized" by machining on the OD of the inner capsule wall.

The computer program that determines the sizes of the gaps was developed at WHC and written in Basic for use on a personal computer. The experimenter must specify the capsule loading as a homogeneous mix per unit capsule length or as an option in smaller 0.5 inch increments. Other inputs include capsule location for determining the local gamma heating rates, local sodium coolant

temperatures, and the desired test temperatures. The computer output provides the required gap and gas mixture for nominal reactor conditions. The computer output is located in the Test Engineering and Development general files. The independent overcheck (Greenslade 1988) is included in Appendix L.

The ability to change the thermal conductivity of the annular gap permits a great latitude for uncertainties in the design parameters and reactor gamma heating rates. The nominal peak gamma heating rate of 4.0 W/gram for a Row 4 position may be in error by 10 percent. This can easily be accommodated by the experiment to maintain the desired temperature. Dimensional changes in the annular gap as a result of irradiation-induced swelling and increases in radiation heat loss as a result of capsule oxidation can also be accommodated within the extent of the control factor range discussed previously.

Materials used to fabricate capsules, support tubes, and other test hardware will be compatible with reactor sodium. All materials will be listed on detailed engineering drawings and will also be included in Volume 1B of the Test Design Description. Materials used to fabricate specimens will be compatible with reactor sodium or will be encapsulated in materials which are compatible with reactor sodium. The composition of the alloy groups used in the specimens will be listed in Volume II of the Test Design Description.

The hardware within Fusion MOTA will be limited to 335 EFPD at 300 MW. The reason for the limitation is the effect of irradiation on the double-walled capsules. The two walls of the capsules will be operating at different temperatures and will have different irradiation induced swelling characteristics. Any change in the insulating gap dimensions within this time will be very small and can easily be accommodated by the gas control system. The capsule walls will be fabricated from 15-30 percent cold worked 316 stainless steel to keep the swelling to a minimum.

6.1.1 U.S. Materials Program Canister Design

The U.S. Materials program will utilize standard gas gapped and weeper canisters as previously discussed.

6.1.2 MONBUSHO

The MONBUSHO program will utilize standard gas gapped and weeper canisters as previously discussed.

6.1.3 BEATRIX-II

As discussed previously, for a fusion reactor to be commercially feasible it must generate its own supply of tritium fuel. One proposal for accomplish-

ing this is to surround the plasma with a lithium ceramic blanket or "solid breeder".

Many solid breeder materials have been investigated recently (Porter 1982; Hollenberg 1984; Hollenberg 1986). Some (LiAlO_2 and Li_2ZrO_3) offer thermal and irradiation stability while others (Li_4SiO_4) offer the possibility of low temperature operation. But Li_2O will be the primary focus for the BEATRIX-II test in Cycle 11. Li_2O possesses by far the best blanket neutronics because of its higher lithium atom density. It is the only solid breeder material that can potentially provide breeding ratios adequate for tritium self-sufficiency without the use of a neutron multiplier. Li_2O does suffer from apparent swelling and the potential of mechanical interaction.

One capsule will be devoted to temperature change experiments while another capsule will be more prototypic of the temperature gradients that exist in an actual blanket. The temperature change experiment will use a ring specimen geometry. The purpose of the temperature change experiment, shown in Figure 6, is to make stepwise changes in temperature and monitor the kinetics of tritium release during these changes. The solid pellet capsule, shown in Figure 7, will be operated with heat generation rates and dimensions which will provide a temperature gradient representative of what might exist in an actual fusion blanket. Tritium release from this capsule will also be monitored; however, temperature changes will only be caused by changes in thermal heat transfer in the capsule due to irradiation.

The thermal stability of a solid breeder material during irradiation is considered important since each solid breeder material has a temperature window for optimum performance. Each solid breeder material is expected to have an upper temperature limit resulting from phenomenon such as lithium transport, sintering, creep, etc. The lower temperature limit arises from tritium retention by the material. The thermal conductivity of the solid breeder is to be measured during the irradiation as a function of both temperature and burnup in the temperature step change capsule. In the solid pellet capsule, the overall thermal stability of the solid breeder will be measured which will reflect not only changes in the intrinsic conductivity but also crack formation, material rearrangement and transport, and burnout effects.

6.1.3.1 Temperature Change Capsule. The temperature change capsule (Figure 6) specimen is a ring of Li_2O which is approximately 0.73 inch OD x 0.059 inch wall thickness and 3.5 inches long. The Li_2O is enriched with ^6Li to achieve the desired operating temperatures and end-of-life Li burnup.

The inside of the specimen is lined with a nickel screen which will help to maintain position of the Li_2O material in the event it fractures into smaller pieces during irradiation. The screen also serves to hold the two 3/32 OD, 316 stainless steel sheathed thermocouples which penetrate the bottom of the subcapsule in position close to the specimen. The thermocouples measure specimen inside diameter temperature at two points 180° apart. The thermo

BEATRIX-II RING SPECIMEN TEMPERATURE CHANGE CAPSULE

Figure 6. BEATRIX-II Ring Specimen Temperature Change Capsule

BEATRIX-II SOLID PELLET CAPSULE

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Figure 7. BEATRIX-II Solid Pellet Capsule

couple sheaths are coated with a nickel aluminide tritium permeation barrier inside the specimen subcapsule.

Two sweep gas tubes also penetrate the subcapsule lower end cap. The sweep gas supply tube terminates at the subcapsule lower end cap. The sweep gas return line terminates near the top of the subcapsule. The sweep gas flows from the supply tube, through a diffuser, through the central void in the ring specimen where it mixes with the tritium released from the specimen, and then out the return tube. An Inconel 600 spring and 316 stainless steel pressure plate are located at the top of the subcapsule to maintain the specimen position during capsule assembly and handling. The spring will exhibit less than a 500 gram load on the specimen during irradiation.

All the surfaces (except weld joints) of the specimen subcapsule wall and end caps, internal thermocouples, and sweep gas return lines (OD only) are coated with a tritium permeation barrier. The barrier is diffusion bonded to the 316 stainless steel substrate and will decrease the permeability by at least a factor of 100. The inside diameter of the subcapsule wall is coated with a 0.005 to 0.010 inch thick nickel 270 corrosion barrier following application of the permeation barrier. The estimated permeation losses from the capsule are <1% of the total generated and are discussed in Appendix F.

The specimen subcapsule is sealed within the 316 stainless steel inner capsule. The annulus between the subcapsule and inner capsule is filled with sodium (a gas plenum to allow for thermal expansion is at the top). The sodium provides a high thermal conductivity fluid around the subcapsule to minimize axial temperature gradients. The annulus between the subcapsule and inner wall is necessary to allow clearance for the one 3/32 inch outside diameter (OD) stainless steel sheathed Type K thermocouple which measures the subcapsule OD temperature. The inner capsule is not treated with a permeation barrier. The gas gap steps are machined into the OD of the inner capsule wall to provide the desired temperature and uniform axial gradient.

The outer wall of the capsule assembly is 316 stainless steel and is similar to the standard MOTA outer wall. All three thermocouples, the sweep gas tubes, and the temperature control gas inlet tube penetrate the outer wall near the bottom end cap. The temperature control gas outlet tube penetrates the outer capsule wall near the top end cap. The sweep gas return line is encapsulated within the core region by another tube which is also coated with a permeation barrier. The annulus between the tubes contains a tritium getter to measure the cycle average tritium permeation from the sweep gas return line.

6.1.3.2 Solid Pellet Capsule. The solid pellet capsule (Figure 7) specimen stack is Li_2O which is nominally 0.67 inch OD x 3.5 inches long. The actual pellet diameter is selected to match the heat generation from the ^6Li enrichment necessary to reach the goal Li burnup with the desired operation temperature. The pellet stack includes depleted Li_2O insulator pellets at the top and bottom, solid pellets at the center, and annular pellets between the insulator

and solid pellets. The annular and insulator pellets have a center hole to accommodate the nominal 3/32 inch OD thermocouple. The ^6Li enrichment of the pellet stack between the insulators will be the same throughout.

Two thermocouples will measure the interior temperature of the pellet stack. The design operating temperature at the beginning of irradiation is 900°C. The thermocouples are 0.068 inch OD W-22Re sheathed Type K similar in design to those used for SP-100 MOTA. The portion of the probe within the pellet stack will be protected by a 0.01 inch thick Inconel 600 thermowell/corrosion barrier which is plated with 0.001-0.002 inch thick nickel. The thermocouple transition sleeves are brazed into the top and bottom end caps.

The capsule wall and end caps are 316 stainless steel. The outer surface of the capsule wall and the inner and outer surfaces of the end caps are coated with the same tritium barrier discussed for the temperature change capsule. The inner surface of the capsule wall will be oxidized and then coated with a 0.04 inch thick layer of porous nickel. The nickel will be approximately 50 percent of theoretical density and will have a filtration rating greater than 10 microns. The purpose of the porous nickel layer is to provide a flow path for the sweep gas around the pellet outside surface and still maintain good thermal contact between the pellet and outer wall.

The sweep gas will enter the capsule bottom, flow up through the porous layer surrounding the Li_2O breeder material, and then flow out the sweep gas outlet tube at the top of the capsule. The support plate and thermowell at the bottom of the pellet stack and the solid pellets at the center of the stack are to minimize sweep gas flow through the "hot" center portion of the pellet stack and thereby minimize Li transport.

The sweep gas lines are 3/32 inch OD x 0.02 inch wall thickness 316 stainless steel. The tubing containing the sweep gas exiting the capsule is also treated with a permeation barrier. Permeation losses from this capsule are <1% of the total generated and are also discussed in Appendix F.

6.2 IRRADIATION CONDITIONS

To achieve the objectives of the experiment, Fusion MOTA must be located in Row 4, position 2406. The assembly will be irradiated for a maximum of 335 equivalent full power days (@ 291 MW) before the test hardware is replaced.

The nominal irradiation temperatures will range between 370 and 900°C. The test temperature for each capsule will be a function of its position in the reactor, specimen contents, and thermal insulation from the reactor coolant.

The bulk mixed mean exit (at 66 inches above core midplane) sodium temperature will be $454 \pm 10^\circ\text{C}$. The test assembly is designed to operate with

a sodium flow of $14,000 \pm 2000$ lbs/hr. The inlet temperature should be $360 \pm 10^\circ\text{C}$.

6.3 INSTRUMENTATION AND CONTROL SYSTEMS

6.3.1 Temperature I&C System

6.3.1.1 General Principles of Operation. The requirements for the Fusion MOTA control and data acquisition system are numerous, but the main necessities are listed below:

- Storage of temperature information from each of the thermocouples.
- Control of temperatures by controlling the percentages of helium and argon gas in the gas gaps.
- Ability to interact with an operator for warnings and remote override capabilities.

6.3.1.2 Control Mode Theory. Each of the capsules whose temperature is controlled by variations in the helium/argon gas mixture will have a temperature control band between a normal high temperature (TNH) and a normal low temperature (TNL) (Figure 8). As long as a capsule target temperature (TN) does not exceed TNH or drop below TNL, the gas mixture in that capsule remains constant. If TNH is exceeded, the computer will calculate and blend a new gas mixture and distribute this gas via the head compartment manifold to the capsule.

In the case of TNH being exceeded, the new blend will have a higher percentage of the more conductive helium. If the capsule temperature drops below TNL, a new blend with a higher percentage of argon gas will be routed to that capsule. If the purge temperature (T_p) is exceeded, a temperature alarm signal will be activated on the control console and in the reactor control room. The computer will then bypass the blender with helium and route it to the capsule. In addition to this "automatic helium purge", a manual "emergency helium purge" capability exists in case of computer or multiplexer failure.

For the BEATRIX-II temperature change capsule, the control band will be reduced to $\pm 2^\circ\text{C}$. The method of operation will be modified to minimize the amount of blend activity. The provision for an automatic helium purge of the BEATRIX-II temperature change capsule will be investigated since this feature may not be desirable for this experiment.

6.3.1.3 Automatic Control Mode Operation. During irradiation, thermocouples will measure temperatures in all of the capsules. Each capsule will have a target irradiation temperature selected by the experimenter. If the measured

BSME/MOTA TEMPERATURE CONTROL

PRIORITY	PARAMETER	CPU FUNCTION	TLS FUNCTION
1	TEMP (T) > PURGE TEMP (T_P)	PURGE (INTERRUPT BLENDS)	TEMP ALARM AND PRINT OUT
2	TEMP (T) > NORMAL HIGH (T_{NH})	BLEND AND DISTRIBUTE	PRINT OUT
3	TEMP (T) < NORMAL LOW (T_{NL})	BLEND AND DISTRIBUTE	PRINT OUT

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Figure 8. MOTA Temperature Control Responses

temperature deviates from the target temperature beyond an allowed amount (T_{NL} or T_{NH}), the control system will respond with corrective action. The control system will determine a new gas blend with the necessary thermal conductivity to return the test temperature to that specified by the experimenter. The blender will be commanded by the control system to mix the new gas blend. A vent and output valve will be opened at the manifold and remain open until the blender "comes to steady-state" and has flushed the old gas mixture from the gas supply line connecting the blender and manifold. Next, the control system will open the proper manifold valve to the capsule and close the manifold vent. Gas will continue to flow from the blender until the capsule is completely flushed with the new mixture. Then, the valve will close and the control system "will wait" until the capsule reaches a new steady-state temperature. The process will be repeated if the temperature remains outside the limits T_{NL} or T_{NH} .

6.3.1.4 **System Components.** The temperature control system major components are listed in Table 7 and are shown graphically in Figure 9.

Table 7
MOTA INSTRUMENTATION AND CONTROL SYSTEM COMPONENTS

<u>Item</u>	<u>Quantity</u>	<u>Location</u>
Gas supply	Three argon bottles Three helium bottles (shared with LMR MOTA)	Operating deck (550 foot level)
Helium protection system	One 19-inch instrument rack (redundant sensors)	Operating deck
Gas blenders	One plus an auxiliary	Operating deck
Computer controls and data acquisition (including manual control and temperature limit system)	Two standard 19-inch instrument racks	Operating deck
Gas distribution	One 19-inch instrument rack	Head compartment
Manual valve switches and display	One 19-inch instrument rack	Operating deck
Alarm, relay, and blender output panel	One 19-inch instrument rack	Operating deck

FUSION MOTA TEMPERATURE INSTRUMENTATION AND CONTROL SYSTEM

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Figure 9. Fusion MOTA Temperature Instrumentation and Control System.

6.3.1.5 System Description

6.3.1.5.1 Gas Supply. Three argon and three helium gas bottles each with 220 cubic feet @ STP capacity are located on the reactor operating deck to provide gas to the blenders. Inlet pressure to the blenders is regulated to approximately 140 psi. The argon/helium gas supply will be shared with the LMR MOTA system (Bauer 1987).

6.3.1.5.2 MOTA Helium Protection System (MHPS). To eliminate the possibility that any gas other than helium can be supplied to the helium input of the temperature control system, each helium bottle is automatically analyzed when it is connected to the gas supply manifold. A thermal conductivity analyzer measures the helium content of the gas, and a computer opens the solenoid valves between the bottle and the control system if the helium purity is > 98 percent. The MHPS is a secondary check to ensure only helium is connected to the system. The primary check remains the administrative control of the bottle installation.

In the event the MHPS will not function, manual bypass valves can be opened with the proper authorization. The MHPS is shown schematically in Figure 10.

6.3.1.5.3 Gas Blenders. The MOTA blender will accept argon and helium from the primary and backup supply bottles. An internal regulation system (Figure 11) will automatically switch to the backup gas bottle when pressure from the primary supply bottles drop below a set point. This action will constitute an alarm condition which will activate the local and control room alarms. These alarms are to notify FFTF Operations that the primary gas bottles need to be replaced.

Gas is blended through mass flow control valves (Figure 11) and then flows to the distribution manifold through a 0.25 inch OD gas line. For automatic operation, the control valves are controlled from the computer system. Manual control is achieved by setting gas ratio potentiometers on the front of the blender and manually selecting valves at the mimic valve panel.

The auxiliary blender is identical to the primary blender and is connected in parallel with the primary blender. Thus, the auxiliary blender can provide gas mixtures under computer or manual control in the event the primary blender is inoperative. In the case of a blender default, the switch from primary to auxiliary blend will occur automatically.

6.3.1.5.4 Computer Controls and Data Acquisition. The heart of the MOTA instrumentation and control system is two HP-9826 computers (Systems A and B) which perform the following functions:

- Acquire and record data from thermocouples.

MOTA HELIUM PROTECTION SYSTEM

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Figure 10. MOTA Helium Protection System Schematic

Temperature Control System Blender Piping Schematic

Figure 11. Temperature Control System Blender Piping Schematic

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- Calculate gas mixtures based on temperature data.
- Provide automatic distribution of calculated gas mixture to specified canister.
- Provide control signals to the distribution manifold valves and sense manifold valve status.
- Provide interactive communication with an operator.
- Perform data reduction and display for historical temperature and gas blender data.
- Identify and activate alarm signals such as electronic component failures and temperature limit excursions.
- Provide temperature limit system (TLS) and manual control functions.
- Provide automated helium purge of selected canisters.

Systems A and B are identical in structure but one system is on line controlling temperatures and logging data while the other monitors and alarms out-of-range temperatures. If one system fails or is removed for maintenance, the other system assumes automatic control of blenders and gas distribution (plus temperature limit function) until the other system is brought back on line.

Thermocouple data acquisition, thermocouple analog to digital conversion, solenoid valve relay control, and blender potentiometer control are accomplished by HP-3497 data acquisition/control units which interface directly with the computers. The thermocouples are routed via Type K extension cable to a patch panel within the control system cabinet on the operating deck.

6.3.1.5.5 Manual Control. Manual control can be performed only if no system controller is in automatic. Either Systems A or B can be used as a manual controller.

The manual control performs the following functions:

- Asks operator for canister to be blended.
- Calculates and displays helium percent of blend.
- Prompts operator to set blender servos.
- Prompts operator through blend evolution.
- Displays temperature data (CRT).

- All valves are operated via manual switches at the mimic valve panel.

6.3.1.5.6 Temperature Limit System (Off-Line System). The temperature limit system (TLS) receives temperature data in parallel with the auto (on-line system) and scans and displays this data.

If the temperature of any capsule is greater than TNH or Tp, or less than TNL, the TLS prints the canister designation, time, and temperature on the TLS printer. In addition, temperatures greater than Tp are alarmed by the TLS on the local alarm panel and in the control room. Thus, TLS acts as a backup to the auto (on-line system) for recording out-of-limit temperatures and providing an alarm to reactor operators.

6.3.1.5.7 Alarm Panel. The alarm panel (AP) incorporates an annunciator, alarm lights, and acknowledge switches. The alarm lights consist of the following:

- Canister temp. > Tp
- System A failure
- System B failure
- System alarm
- Main blender helium low
- Main blender argon low
- Aux. blender helium low
- Aux. blender argon low

The temperature out-of-limit light is activated by the TLS in the event a capsule temperature is greater than Tp. This system can also detect open thermocouples. The system alarm light is activated whenever a functional or system component failure occurs. Failures might include data acquisition system failure, valve failure, or data storage failure.

6.3.1.5.8 Emergency Purge Switch and Manual Valve Control Panel. There is an emergency purge switch for the main blender and the auxiliary blender. Pressing this switch will cause the following:

- Close blender mass flow control valves
- Open blender helium bypass valve

- Close manifold vent valve
- Open all manifold capsule supply valves (all capsules will be flooded with helium)
- Deactivate automatic control of MOTA

As discussed previously, this feature may be deactivated for the BEATRIX-II temperature change capsule.

All MOTA system valves and flow paths are represented in mimic form by LEDs. Switches are provided for operation of each system valve for manual control.

6.3.1.5.9 Gas Distribution Manifold. This system is located in the head compartment. A 1/4 inch OD stainless steel tube connects the blender output located on the operating deck with the distribution manifold. Each temperature controlled canister is connected to the distribution manifold by two solenoid valves and a manual valve plumbed in parallel. The solenoid valves are used under normal automatic and manual operation. The manual valve is used in the event that the solenoid valves cannot be operated and control gas must be distributed to the capsule.

The distribution manifold also contains a vent valve which is used to flush the manifold prior to each blend distribution. The vent valve discharges to the reactor containment ventilation system.

6.3.1.6 Safety and Reliability. To ensure against an inadvertent equipment failure jeopardizing test specimens and data, the instrumentation and control system was designed with backup redundancy as a prime criteria. This redundancy provides a high level of confidence that the capsule temperatures can be maintained at or below the target temperatures. The backup components are listed below:

6.3.1.6.1 Gas Supply System. Backup gas bottles of argon and helium will automatically provide gas to the blenders whenever the primary bottle pressures drop below a given set point. This will also allow changing gas bottles without interrupting blending, distributing, or purging operations.

6.3.1.6.2 MOTA Helium Protection System. This system incorporates redundant thermal conductivity analyzers and solenoid valves as well as the manual bypass valve discussed previously.

6.3.1.6.3 Gas Blenders. An auxiliary blender is installed into the system and can be computer controlled to provide gas to MOTA in the event that the main blender is inoperable. In addition, the helium supply solenoid valve and the helium purge solenoid valve have parallel hand valves. This provides a

method of purging with helium if a valve fails or if power is lost to the blender.

6.3.1.6.4 Computer Controls and Data Acquisition. Computer controls and the data acquisition system will employ two systems with the primary system, backed up by the secondary. If one fails, the redundant system can take control of gas blending, distribution, and purging as well as data acquisition previously performed by the other. It will also provide a "system" alarm signal to the alarm panel to alert operations personnel of a malfunction. TLS functions are also picked up by the redundant system when one system has failed.

Electronic components used to interact with the main blender will also parallel the auxiliary blender so a single component failure will not preclude control of the MOTA gas mixtures.

6.3.1.6.5 Manual Control/Temperature Limit System/Alarm Panel Manual Control. In the manual mode, valves are switched at the mimic panel (except servo valves which must be set on the blender front panels). If Systems A and B functions fail, a helium purge can be activated by depressing the PURGE button on the mimic valve panel. These signals remove power from the blender valves which are designed to "fail closed", thus assuring no argon can be discharged into the MOTA capsules. The signals also bypass the gas distribution valve signals and are wired directly to the manifold chassis to open all distribution valves and close the vent valves.

6.3.1.6.6 Temperature Limit System. The TLS is connected in parallel with the auto (on-line) system to monitor capsule thermocouples. The TLS is independent of the on-line system except when it periodically receives updated system parameters from the on-line system. A TLS failure will be detected by the auto system and annunciated on the alarm panel as a "system failure".

The TLS will record and print any capsule temperature that is greater than T_p and will provide a "temperature alarm" signal to the alarm panel. If the auto system fails, the TLS will assume the auto control function plus TLS.

6.3.1.6.7 Gas Distribution and Signal Conditioning. The distribution manifold has a primary and secondary valve in parallel to supply gas to each capsule. The primary valves are driven by one DAS and the secondary valves by a separate DAS. All valves interface with Systems A and B controllers and mimic panel. Thus if a DAS fails, control of one capsule gas supply valve is assured from any of the three control sources. A manual valve is also plumbed in parallel with the solenoid valves as previously discussed.

6.3.2 BEATRIX-II Self-Powered Neutron Detector (SPND)

One self-powered neutron detector (SPNDs) will be located adjacent to the BEATRIX-II vented capsules in Level 1. The purpose of the SPND is to monitor the relative reactor power over the course of Cycle 11 operation in the vicinity of the specimens. The capsules will also contain dosimetry for determining the cycle average neutron spectrum.

The SPND will have a signal conditioner located in the head compartment. The conditioned signals will be displayed and recorded by the tritium handling system instrumentation and control system.

6.3.3 BEATRIX-II Tritium Handling System

The supporting systems for the BEATRIX-II tritium handling system are listed below:

- Sweep gas inlet system
- Sweep gas analysis system
- Tritium recovery system
- Tritium data acquisition system

The arrangement of these systems is shown in Figure 12. The design features and requirements for each system are described in the following sections.

6.3.3.1 Sweep Gas Inlet System. The sweep gas inlet system is made up of two parts:

- Sweep Gas Inlet Monitoring System
- Sweep Gas Inlet Pressure Control System

The Sweep Gas Inlet Monitoring System is shown schematically in Figure 13. It contains the bottled gas supply, purification, and purity analysis instrumentation. The design pressure for the Inlet Monitoring System is 150 psig. (The only exception is the section with the oxygen analyzer which is rated only for 7 psig.)

Beatrix-II Tritium Handling System and Temperature Control System Arrangement

Figure 12. BEATRIX-II Tritium Handling System and Temperature Control System Arrangement

Beatrix-II Sweep Gas Inlet Monitoring System

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Figure 13. BEATRIX-II Sweep Gas Inlet Monitoring System

Sweep gas will be supplied in pressurized cylinders containing approximately 200 cubic feet of gas. The cylinders will be connected to two manifolds so that either a reference gas or an alternate gas may be selected. The reference gas will be helium with 0.1 percent hydrogen or deuterium. The alternate gas will be helium with 0-1 percent hydrogen or deuterium. Gas from the cylinders will flow through a gas purifier unit to remove oxygen and water to less than 1 ppm. An analyzer will be used to measure oxygen content of the sweep gas prior to routing the gas to the BEATRIX-II capsules each time a new bottle is installed on the manifold.

After it has passed through the inlet monitoring system, the sweep gas enters the Inlet Pressure Control System (Figure 14). The design pressure for the Inlet Pressure Control System is 150 psig. The gas flows through a gas operated isolation valve and then into moisture meters. After leaving the moisture analyzers, the sweep gas is directed to individual controllers and pressure gauges and then to the BEATRIX-II capsules.

The reactor pressure boundary is marked by an ASME Boiler and Pressure Vessel Code Section III class 1 valve at the outlet of the inlet pressure control system. The sweep gas tubing from this valve to the MOTA Instrumentation Penetration System Bulkhead forms the reactor pressure boundary.

The Sweep Gas Inlet System is constructed of 1/4" O.D. x 0.035 wall stainless steel tubing. The tubing downstream of the class 1 valve is 1/8" O.D. x 0.035" wall 316 stainless steel. Connections are made using either welding or Cajon VCR connectors. The valves other than the class 1 pressure boundary valves are Nupro manually operated bellows sealed valves with Vespel (polymide) seats. Bypass lines are included in the design for maintenance or replacement of individual components.

6.3.3.2 Sweep Gas Analysis System. Sweep gas exiting the Fusion MOTA test assembly will flow through 1/8 inch outside diameter x 0.035 inch wall thickness stainless steel tubes to a stainless steel glove box which will contain the components for the sweep gas analysis and tritium recovery systems. An ASME Boiler and Pressure Vessel Code Section III class 1 valve marks the outlet side of the reactor pressure boundary and is positioned at the point just before the sweep gas tube penetrates the glovebox. The internal surfaces of the stainless steel tubing and fittings will be cleaned per TDD-1B to improve the performance for tritium handling.

The glove box dimensions are 30 inches wide by 108 inches high by 84 inches long (157.5 cubic feet). The glove box has double-sided access and an argon atmosphere. Automatic pressure control of the glove box will compensate for the large variations in the FFTF containment pressure.

Beatrix-II Sweep Gas Inlet Pressure Control System

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38804-063.4M

Figure 14. BEATRIX-II Sweep Gas Inlet Pressure Control System

The sweep gas analysis system (Figure 15) will provide on-line continuous analysis of the gas and a sampling port for periodic single sample collection. The design pressure for the sweep gas analysis system is 150 psig. Instrumentation in the glove box will measure total tritium concentration, pressure, flow rate, and water concentration in the sweep gas. The flow stream first passes through a moisture meter and is then split in half by flow controllers. One half of the stream flows through a molecular sieve which removes the moisture, through another moisture meter, and then into the ion chamber. This ion chamber measures the fraction of tritium which is gaseous. The second half of the flow stream passes through an electrolysis cell which converts the moisture to gaseous hydrogen, then through a moisture meter, and then into another ion chamber. This ion chamber measures the total tritium concentration in the flow stream.

The two flow streams then converge and flow through a pressure gauge/controller to the Tritium Recovery System.

Tritium concentration in the glove box atmosphere will also be monitored. Stainless steel tubing, packless diaphragm or bellows type valves, and Cajon VCR metal gasket fittings will be used to fabricate the system. Separate analysis systems as shown in Figure 15 will be installed for the temperature change and solid pellet capsules.

Tritium will be measured with a four decade ion chamber. The lowest range will be 0-10 Ci/m³ and the highest range will be 0-10,000 Ci/m³. Water concentration will be measured from 0-100 ppm with a ± 2 ppm accuracy.

A separate helium source is supplied to flush the majority of tritium from the lines prior to breaking connections for either maintenance or sampling. The gas sampler is positioned such that a sample can be taken from any point in either flow stream ahead of the ion chambers.

6.3.3.3 Tritium Recovery System. The tritium recovery system shown schematically in Figure 16 will remove tritium and hydrogen/deuterium from the BEATRIX-II sweep gas effluent flow stream. The design pressure for the tritium recovery system is 150 psig. The tritium recovery process will utilize metallic getter materials. The tritium will be immobilized in units less than 1000 Ci each to facilitate shipping and waste disposal.

The sweep gas streams from the temperature change and solid pellet capsules will be mixed and then flow through an ion chamber to measure the total tritium concentration. The gas will then pass through one of two parallel primary getter beds. The tritium concentration will be reduced by a factor of 2500 as it passes through the getter. The exit concentration will be measured by another ion chamber. A secondary getter bed is positioned downstream of this ion chamber in the event the primary bed fails or becomes saturated. The tritium load on the primary bed will be calculated by the

Beatrix-II Tritium Measurement and Gas Analysis System

38804-063.3M

Figure 15. BEATRIX-II Tritium Measurement and Gas Analysis System (typical for all monitored capsules)

Beatrix-II Tritium Recovery System

Figure 16. BEATRIX-II Tritium Recovery System

Tritium Data Acquisition System using the Analysis System Flowmeter and Tritium Recovery System inlet ion chamber data.

The sweep gas then flows through a pressure controller which maintains the tritium recovery system at a constant pressure which is greater than the anticipated highest pressure of CAPS. It then passes through an isolation valve and into CAPS.

Secondary containment for the tritium recovery system components will be provided by the stainless steel glove box described in the previous section. A second recovery system will remove tritium which may be released to the glovebox atmosphere during maintenance or sampling. A refrigerated cooling unit rated at 10,000 BTU/hr will remove heat which is generated by process components inside of the glove box.

6.3.3.4 Tritium Data Acquisition System. The tritium data acquisition system is separate from the temperature control system. The purpose of the I&C system is to monitor the BEATRIX-II instrumentation and to record and display the data. The system will also generate and display the BEATRIX-II system alarms. The system is also responsible for tracking the tritium load on the recovery system getter beds.

Data will be recorded on magnetic disk for later analysis with selected data displayed on chart recorders. Inputs to the tritium data acquisition system from the sweep gas inlet, analysis, and tritium recovery systems will include the following:

- Pressure gauges
- Flow meters
- Ion chambers
- Process thermocouples
- H₂O monitors
- Self-powered neutron detectors
- Canister thermocouples (in parallel with temperature I&C system)

6.3.3.5 Safety and Reliability. To ensure against an inadvertent equipment failure jeopardizing data and safe operation of the system, many redundant features are built into the tritium handling system. The backup components for the systems are as follows:

6.3.3.5.1 Sweep Gas Inlet Monitoring System. Three separate bottles are included in both the reference and alternate gas supplies. These bottles are installed such that any bottle may be replaced while the others remain on line. A pressure relief valve is included to prevent overpressurization of any of the system components (150 psig). Redundant oxygen meters are provided for measuring the oxygen contamination in the sweep gas prior to routing it to the capsule.

6.3.3.5.2 Sweep Gas Inlet Pressure Control System. Bypass valves are provided so that the sweep gas can be routed around the instrumentation in the event of a failure.

6.3.3.5.3 Tritium Measurement and Gas Analysis System. Bypass valves are provided so that the sweep gas stream can be routed around a failed component while it is replaced. If required, the flow can be shut off entirely through one leg of the system and routed through the other. This would result in a partial loss of data.

6.3.3.5.4 Tritium Recovery System. Redundancy is provided for the primary tritium recovery getter. Only one of the primary getters is on line at a time; therefore, the other primary getter is available to be put on line when the first becomes saturated. A secondary tritium recovery getter is downstream of the primary system and will collect the tritium should the primary system fail.

Redundancy is provided for the inlet tritium meter by the Gas Analysis System ion chambers. Redundancy is provided for the outlet tritium meter by the plant effluent monitor. The plant effluent monitor is the official tritium release measurement device. The tritium meters within the Tritium Handling System are only for experimental data acquisition and process control.

6.3.3.5.5 Tritium Data Acquisition System. Reliability is provided by redundant hardware. The system is designed similar to the temperature control system in that there are dual computers and data acquisition system. Only one of the systems is on line at a time while the off line system monitors for failure of the on line system.

6.3.3.5.6 Tritium Handling System Glovebox. The glovebox provides secondary containment of the tritium should any be released by failure of a system component, during maintenance, or sampling. Overpressure protection is provided by a redundant automatic pressure control system which is backed up by a bubbler type pressure relief system developed by Los Alamos National Laboratory for tritium glovebox operations at the Tritium Systems Test Assembly.

7.0 TEST EVALUATIONS

7.1 THERMAL ANALYSIS

The thermal analysis covers the temperature distribution in the capsule only. The test vehicle and capsule support structure were not included since they will be in intimate contact with the reactor coolant. The temperatures of these components are expected to be near that of the local coolant and will not be subjected to significant thermal stresses. The test specimens were considered only from the aspect of how they would influence the temperatures in the capsule. The intention of the analysis was to demonstrate the integrity and operability of a capsule under worst case conditions.

The analysis considered only the nominal 1300°F experiment with a maximum specimen loading of 25 percent steel (by volume) of the specimen region. This is the worst case for heat generation and maximum temperatures. Full power operation of 4 watts/gm maximum gamma heating was used. A 25 percent overpower operating condition was analyzed to demonstrate the operational safety of the capsule. The maximum gamma heating rate of 4.0 watts/gram was originally thought to be conservative. The gamma heating rate listed in Appendix E of the User's Guide for a nonfueled open test assembly in a row 4 position is 3.38 watts/gram.

HM032 (MOTA-1A) operating experience in Cycles 2 and 3 at 400 MW has indicated that the nuclear heating in stainless steel is approximately 50-60 percent higher than the design value of 3.39 W/g. The major impact of the increased heat deposition is the higher percentages of helium that must be used to maintain design temperatures. Despite the increased heat, the MOTA I&C system has satisfactorily controlled temperatures well below the boiling point of sodium (883°C at 1 atm).

The input conditions for the analysis were:

- Ambient sodium coolant temperature of 690°F.
- Conductivity of the gap gas midrange between pure helium and pure argon.
- Gamma heating rate of 4.0 watts/gram.
- Uniform gamma heating field over the entire capsule.
- The complete analytic report is in Appendix A of this document.

Although this analysis is not as conservative as originally thought due to the higher than predicted nuclear heating, the calculation of the thermal stresses are still valid. The case analyzed is actually the worst normal operating case.

The worst off normal case was actually encountered in the operation of MOTA-1D during FFTF Cycle 7 (Ermi 1986) when a bottle of argon gas was inadvertently connected to the helium supply of the temperature control system. The result was that all 25 temperature controlled capsules were filled with pure argon and the temperatures reached 1010°C (1850°F) for 50 minutes. Following the event, all of the capsules were able to return to normal operation which indicates the capsules did not fail under this worst case condition. Therefore, the capsules are structurally sound under all operating conditions.

7.1.1 U.S. Materials Program

The analysis described above was in support of the original design of the MOTA. Operation of the MOTA and additional thermal analysis have shown that temperatures at the ends of the specimen region were in excess of those desired (Bauer 1984; Bauer 1986). For this reason, the capsule design has been changed to "flatten" the temperature gradient. Modifications to the design include the following:

- Shortening the length of the large insulating gap at the top of the capsule.
- Addition of a shim at the sodium inlet end of the capsule.

The capsule gas gaps are determined in the following manner:

- The experimenter identifies the capsule operating temperature, position within the MOTA, and nominal specimen loading.
- A one-dimensional (radial) heat transfer design code is used to determine gas gap dimensions in 0.471 inch steps along the specimen compartment.
- Another one-dimensional (radial) heat transfer design code is used to determine gas gap dimensions for the shim at the sodium inlet end of the capsule.
- The gas gap dimensions are overchecked by an independent party.

The shim is necessary to compensate for the heat loss due to sodium flowing into the capsule. Because capsules typically operate at different temperatures, as well as at different gamma heating rates, the shim must be custom-sized for each capsule.

The calculation of the shim size requires the sodium flow rate through the capsule to be known. The pressure drop across the capsule is known from a full scale water flow test as discussed in Appendix I. The flow through an individual capsule is estimated from water flow data taken from a capsule mockup. This testing is described in Appendix G.

A two-dimensional thermal overcheck was performed for the LMR MOTA-1F design and the results are included in Appendix C. The overcheck concluded that a slight gradient existed at the top of the canister; however, the remainder of the specimen region is essentially isothermal. The top of the canister was subsequently further modified to eliminate the gradient. The complete heat transfer analysis (including computer runs) are in the general file for this experiment under the cognizance of Test Engineering and Development.

7.1.2 MONBUSHO

The MONBUSHO capsule thermal analysis is identical to that performed for the U.S. Material program as discussed in section 7.1.1.

7.1.3 BEATRIX-II

Thermal analysis of the BEATRIX-II ring specimen temperature change capsule and the solid pellet capsule is required to verify that the test objectives can be obtained.

7.1.3.1 Ring Specimen Temperature Change. A one-dimensional (radial) heat transfer performance code similar to that used for the U.S. and MONBUSHO capsule design has been developed by WHC. The code predicts the temperatures within the capsule shown in Figure 6. The predicted temperatures and pertinent input parameters for the reference specimen design are included in Table 8.

A two dimensional (r,z) heat transfer code was used to predict the temperature distribution within the capsule. This analysis is included in Appendix H. This code was also used to size the gas gaps at either end of the inner wall to decrease the temperature difference between the specimen midplane and ends to 15°C, which is well below the 25°C requirement.

7.1.3.2 Solid Pellet Capsule. A one-dimensional (radial) heat transfer design code has been developed by WHC to size the pellet diameter necessary to meet the 900°C centerline temperature for the reference capsule design shown in Figure 7. The results for the reference design are shown in Table 9.

Table 8
TEMPERATURE CHANGE CAPSULE THERMAL ANALYSIS
1.5 MM THICK RING SPECIMEN - POSITION 1-E

Specimen Temperature (°C)		Delta-T (°C)	% He
ID	OD		
645	617	28	0
492	468	24	100

Pellet ID = 0.613" = 1.557 cm
 Pellet OD = 0.731" = 1.857 cm
 Heat Generation Rate = 100 w/cc (BOC)
 Gas Gap = 0.005" = 0.012 cm
 Total Li Burnup @
 300 Days = 4% (61% ⁶Li enrichment)
 Thermal Blanket = sodium
 Li₂O Density = 85% T.D.

Table 9
SOLID PELLETS CAPSULE THERMAL ANALYSIS
POSITION 1-C

T_{ID} = 900°C
 T_{OD} = 441°C
 Coolant Temperature = 379°C
 Heat Generation = 100 w/cc (BOC)
 Pellet ID = 0.094" = 0.24 cm
 Pellet OD = 0.670" = 1.702 cm
 Total Li Burnup = 4% (61% ⁶Li enrichment)
 Li₂O Density = 85% T.D.

The same two dimensional heat transfer code was used to predict the temperature distribution in the solid pellet capsule as for the temperature change capsule. The complete analysis is included in Appendix H. The analysis predicts the centerline temperature to be within 25°C of the midplane temperature over 68% of the specimen length. The bottom end is the coldest and is 657°C. This temperature is above the minimum temperature of 600°C specified in section 4.1.21.

7.2 THERMAL HYDRAULIC PERFORMANCE

The coolant temperature rise through the experiment was determined by analysis of thermocouple measurements from previous MOTAs. The Fusion MOTA is essentially identical to the previous MOTAs and is also to be located in a symmetrical row 4 position; therefore, the heat generation and flow rates should be equivalent.

Figure 17 is a plot of MOTA coolant temperature measurements from FFTF Cycles 5 and 9 in position 1406. Based on extrapolation of this data, the coolant temperatures at 66 inches above core midplane (elevation of driver fuel coolant exit) is 454°C (849°F). From data taken from the six assemblies surrounding position 2406 at the beginning of cycle 10A (approximately the same operating conditions as 11A), the average exit temperature of the assemblies is 494°C (921°F) and the maximum exit temperature of any one assembly is 510°C (950°F).

The difference between the Fusion MOTA exit temperature and the average exit temperature of the adjacent assemblies is 72°F (921-849). This is less than the 200°F allowed (Taylor 1978). The difference from the maximum outlet temperature is 101°F (950-849) which is less than the 270°F allowed (Taylor 1978). The 849°F exit temperature is also well below the average and maximum exit temperatures; therefore, the exit temperature obviously does not exceed the average adjacent temperature by the 120°F allowed. Therefore, the flowrate is adequate to meet the requirements of the "User's Guide for the Irradiation of Experiments in the FTR" (Taylor 1978), Appendix I.

Analysis of the increased heat load on the sodium coolant within MOTA using 5.26 W/g (400 MW) and a 125 percent overpower condition indicates a maximum coolant exit temperature of 954°F. This value is well within the limits of 781°F and 1112°F as specified in the FTR User's Guide (Taylor 1978) Figure A-17.

7.3 STRESS ANALYSIS

This analysis covered the steel structural parts of the capsule. The specimens and capsule support structure were not included. Since high stresses were not expected in the support structure, only thermal expansion stresses were evaluated. There are no other significant external loads. The nominal 1300°F case was analyzed at full power operating conditions. The computer model for the stress analysis was the same as the model for the thermal analysis except that the initial dimensions were altered to their room temperature values. The temperature input data was taken directly from the full power thermal analysis. The details of the stress analysis are presented in Appendix A of this document.

MOTA COOLANT TEMPERATURE PROFILE

MHC-SP-0224

Figure 17. MOTA Sodium Coolant Profile

The inlet tube was analyzed separately to determine the stresses resulting from differential expansion of the inner and outer capsule walls. It was determined this differential expansion will result in a maximum stress intensity of 19,436 psi in the inlet tube. The maximum stress intensity in the capsule itself will be 12,267 psi in the cap. Both of these stresses are below the ASME Boiler and Pressure Vessel Code allowable limit of Case 1592-7, Appendix T, Section 1323, December 22, 1975. Since the case analyzed developed the highest expected thermal gradients of any of the intended experiments, it was concluded that all cases will operate without structural failure.

7.3.1 U.S. Materials Program

There is no significant change in the capsule design from previous MOTAs; therefore, additional stress analysis is not required. The changes discussed in Section 7.1 served only to bring the capsule temperature distribution and, hence, stresses within the bounds of the original analysis.

7.3.2 MONBUSHO

There is no significant change in the capsule design from previous MOTAs; therefore, additional stress analysis is not required. The changes discussed in Section 7.1 served only to bring the capsule temperature distribution and, hence, stresses within the bounds of the original analysis.

7.3.3 BEATRIX-II

Stress analysis has been completed for the following components:

- Ring specimen subcapsule temperature change capsule
- Solid pellet capsule

This analysis is documented in Appendix D. No stresses which would result in failure of the capsules were identified.

7.4 NEUTRONICS PREDICTIONS

7.4.1 U.S. Materials Program

The gamma heating and neutron flux for the positions of the U.S. material specimens are well characterized by the 9 cycles of operating experience with MOTA; therefore, no further neutronics predictions are required.

7.4.2 MONBUSHO

The gamma heating and neutron flux for the positions of the MONBUSHO material specimens are well characterized by the 9 cycles of operating experience with MOTA; therefore, no further neutronics predictions are required.

7.4.3 BEATRIX-II

Predictions of the nuclear heating and tritium production rates in the Li_2O breeder material are necessary in order to design the capsule and tritium handling system.

Monte Carlo calculations using a detailed model of the Fusion MOTA below core and Level 1 positions have been completed. The nuclear heating and tritium generation rates predicted by this calculation are used in the design of the capsules. The results of these calculations are presented in Appendix E.

7.5 ORIFICE ANALYSIS

An analysis was performed to properly size the orifice plates in the shield inlet assembly. This analysis determined that the orifice plates should be sized for a flow rate of 14,000 lbs/hour (Cramer 1987).

7.6 ANALYSIS OF CHECK VALVES

Each temperature control gas line which will penetrate the reactor pressure boundary will have a check valve which has been qualified to the intent of the ASME Boiler and Pressure Vessel Code. These valves are modified from standard Swagelok quick connects. The analysis of the check valve is included in Appendix B of this document.

7.7 UNCERTAINTIES, INCLUDING ENGINEERING DEVELOPMENT TESTING

The BEATRIX-II experiment is the first experiment of its kind in FFTF. The 3σ uncertainty in the nuclear heating and tritium generation rates is 42%.

Applying this uncertainty to the solid pellet capsule, the average cladding temperature is 425°C compared to 412°C for the design case. The increase in the cladding temperature is insignificant and will not be detrimental to the capsule performance.

Using the one dimensional heat transfer code described in Section 7.1.3, the solid pellet capsule maximum centerline temperature at this uncertainty is calculated to be 1208°C and the temperature at the thermocouple is 954°C. These values are well below the 1427°C melting point of Li_2O . Both the thermocouple and the specimen would remain intact under these conditions.

Applying the uncertainty to the temperature change capsule also results in higher temperatures; however, failure of the capsule and release of material to the coolant is not predicted.

In both the solid pellet and temperature change capsule, the increased temperature is not a safety concern; however, if the heating rate is at the upper 3 σ limit, it may compromise the test data.

An increase in 42% in tritium production rate from 26 Ci/day to 37 Ci/day will not be detrimental to the safety of the experiment. The safety analysis results discussed in Section 8 are sufficiently conservative to account for this potential increase.

8.0 SAFETY ANALYSIS

8.1 NORMAL OPERATION

8.1.1 Fuel Movement or Restructuring of Test Materials

This is a nonfueled test. All test specimens will be structural metals or ceramics which are inert in a sodium environment.

8.1.2 Changing Heat Transfer Interfaces (e.g., Loss of Sodium Bond Between Fuel and Cladding)

The BEATRIX-II temperature change capsule is the only sodium bonded capsule in the Fusion MOTA-2A. Loss of the sodium in the annulus surrounding the specimen subcapsule would result in overheating of the specimen and subcapsule; however, the outer wall would remain intact in spite of this event. No material would be released to the coolant; however, the experimental data would be compromised.

8.1.3 Changes in Physical and Mechanical Properties of Cladding or Capsule Materials

All outer capsule structural components will be fabricated from 20% CW 316 stainless steel. The basic configuration of the outer capsule and its instrument leads is identical to the standard MOTA capsule (Greenslade 1984).

8.1.4 Reduction of Coolant Flow Caused by Component Swelling, Creep, Material Transport, etc.

All coolant flow paths will be relatively large compared to potential swelling, creep, and material transport (Greenslade 1984).

8.1.5 Stress Corrosion Due to Reactor Environment, Interim Examination, Sodium Removal, etc.

All outer capsule materials are chemically compatible with sodium. Stress corrosion of MOTA structural components has not been observed in previous MOTAs and is not expected for Fusion MOTA because of the same configuration and operating conditions. Each MOTA is constructed of new material and has not been subjected to sodium wetting prior to insertion. Only the previously irradiated test specimens have gone through sodium removal and they are contained within the MOTA capsules. Therefore, failure of the specimens due to stress corrosion is of no safety concern.

8.1.6 Flow Induced Vibration or Erosion Effects

The flow velocity of 6 fps in the duct will be well below the 40 fps required to cause vibration in pipe flow. The space between the capsules and duct walls will be filled with sodium which will dampen free vibrations. Water flow tests have been conducted with a full scale prototype. View ports were included in the test duct and no detectable vibration was observed (Appendix I). No damage due to vibration has been observed in previous MOTAs.

Material erosion will not be a problem since all parts and specimens in the experiment are structural metals and the flow velocities are low.

8.1.7 A Deviation of Power and/or Flow Caused by Misloading the Experiment

Misloading the experiment is extremely unlikely because Fusion MOTA is inserted in an OTA position which is visible to the operators. It would be extremely difficult to operate MOTA in any other OTA position because it would require modification of the instrumentation connector.

Should the assembly be misloaded into a row 2 position, the temperature control system could easily accommodate the conservatively estimated 15% increase in nuclear heating rate. The tritium production rate in the BEATRIX-II canisters would also increase by 15% which could be easily accommodated by the tritium handling system.

8.1.8 Pressurized Specimen Failure

Fusion MOTA will contain a variety of pressurized specimens. Experience with previous MOTAs indicates that failure of single or multiple pressurized specimens within the MOTA canisters poses no threat to safe reactor operation.

8.1.9 Tritium Generation

Tritium generation within the structural materials of U.S. and MONBUSHO canisters is extremely low and will be no different than for previous MOTAs.

The purpose of the BEATRIX-II canisters is to generate tritium. The tritium generated in the nonvented capsules will permeate through the subcapsule walls into the reactor coolant. The predicted tritium generation rate in the nonvented capsules is less than 30 Ci/day at the beginning of Cycle 11. The predicted tritium generation rate in the ring specimen temperature change capsule is 7 Ci/day and 19 Ci/day for the solid pellet capsule at the beginning of Cycle 11. Although under normal operation this tritium will be swept out of the reactor and into the tritium handling system, a conservative analysis assumes all of the tritium permeates into the reactor coolant. The tritium load on the reactor coolant for these conditions is 56 Ci/day which is below the limit of 100 Ci/day (Prevo 1987).

8.2 OFF-NORMAL EVENTS

Off-normal events were evaluated for safety with little or no consideration of the integrity of the experimental data. All off-normal events as outlined in Appendix D of the FTR User's Guide were reviewed. The worst case off-normal event unique to the MOTA test vehicle is an indefinite argon purge of the MOTA canisters. The worst case off-normal event for the tritium handling system is a breach of the sweep gas return lines which is addressed in Section 8.3.12 and Appendix J.

8.2.1 Reactivity Insertion Events

The definition of an unacceptable operating condition is the boiling of sodium. This could commence at 935°C (1715°F) (Fink 1979) at the 1.56 atm pressure at MOTA level 5 and would precede any melting or significant degradation of the stainless steel structural components. All six reactivity insertion events described in the User's Guide, Appendix D, were considered. The

analysis demonstrated that none of the events would precipitate any safety hazards.

The four full power events were considered safe if shutting down the reactor takes less time than heating the steel to 1700°F. The allowable heating time was determined as follows:

Initial conditions:

Reactor power	400 MW
Gamma heating rate	5.5 watts/gm
Sodium flow rate	100 percent
Experiment temperature	1300°F

Event conditions:

Reactor power	500 MW
Gamma heating rate	6.88 watts/gm
Heat removal rate	0.0 BTU/hr (very conservative)

Allowable temperature increase limit:

Max. allowable temperature	1700°F
$\Delta T \text{ max.} = 1700 - 1300 =$	400°F

Allowable heating time at overpower conditions:

$$t = \frac{C_p \Delta T_{\text{max}}}{\gamma} = \text{allowable heating time (seconds)}$$

$$C_p = \text{specific heat of steel} = 0.1424 \frac{\text{BTU}}{\text{lb}} \text{ } ^\circ\text{F at } 1300^\circ\text{F}$$

$$\gamma = \text{gamma heating rate} = 6.88 \text{ watts/gm} = 2.9578 \text{ BTU/lb sec.}$$

$$t = \frac{(0.1424)(400)}{(2.9578)} = 19.26 = 20 \text{ seconds}$$

For all four full power events, the reactor power drops to below 15 percent in much less than 20 seconds (Taylor 1978, Figure D-3). At the same time, the sodium flow is still above 10 percent of full flow (Taylor 1978, Figure D-11). At 15 percent power, 60 MW (safe steady-state power level) and 10 percent flow, the maximum centerline temperature of a worst case capsule in a steady-state condition was calculated to be less than 552°F above the sodium inlet temperature. Assuming the rated inlet temperature of 792°F persisted, the highest temperature in the capsule was 1344°F. This was less than the

limit of 1700°F. Thus, none of the full power events will cause unsafe operating conditions.

The two low power events (Taylor 1978) cause an energy buildup and then a subsidence. The worst case is the 25 cent/sec event which peaks at 490 MW. Integration of the power curve from Table D-IX of the User's Guide shows a heat input of 0.24 BTU/lb from initiation of the event until the power drops below 60 MW. The worst case initial conditions will occur after the experiment has been irradiated sufficiently to produce residual heating in the structural materials. Analysis indicated the maximum specimen temperatures were approximately 790°F. The heat input per pound to raise the temperature of steel from the initial temperature of 790°F to 1700°F is:

$$Q = C_p \Delta T$$

where: C_p = specific heat of steel = 0.1355 BTU/lb at 1000°F
 ΔT = 1700°F - 790°F = 910°F
 Q = 0.1355 x 910 = 123 BTU/lb

This shows that the input of 0.24 BTU/lb during the event will be much less than the 123 BTU/lb limit. Thus, neither of the lower power events will cause any safety hazards.

An analysis of the potential reactivity effects due to release of Fusion MOTA materials to the reactor coolant is included in "Potential Reactivity Effects Due to Release of Fusion MOTA Materials" (Nelson 1988), Appendix J. The conclusion of the analysis is that the maximum 0.12\$ reactivity increase is well within the envelope of reactivity changes determined in FSAR analyses to be benign.

8.2.2 Heat Removal Reduction

Loss of electric power or failure of a primary pump would cause loss of coolant flow followed by a reactor scram. The power and flow transients applicable early in the experiment for these events are shown in the User's Guide, Appendix D, in Figures D-1, D-12, D-13, and D-14. In both cases, the power would be reduced to below the tolerable 15 percent level in less time than it would take to overheat the worst case capsule. Neither event would cause a safety hazard.

Later in the experiment, the structural materials will have received sufficient irradiation to produce residual gamma heating during and after a reactor scram. This condition was evaluated using the residual heating curve for structural materials from the Guide for Irradiation Experiments in EBR-II. The worst case loss of flow was the mechanical failure of a primary pump (E-1

event). A transient thermal analysis of a worst case capsule indicated that no safety hazard would result. Details of this analysis are in Appendix A of this document.

8.2.3 Overpower Operation

Analysis of a 1300°F capsule in a steady-state condition of 125 percent of the rated reactor power indicated that the maximum temperature which occurs will be 1524°F. This is below the boiling point of sodium and will not significantly degrade the structural materials in the capsule. Thus, no safety hazard will result from the 125 percent overpower transient. The tritium generation rate in the BEATRIX-II vented capsules would be increased from 26 Ci/day to 32.5 Ci/day. This increase could be easily accommodated by the Tritium Handling System and would result in no safety hazard.

8.3 FAILURE ANALYSIS

8.3.1 Gas Blanketing

Gas blanketing of the capsules would greatly reduce the heat removal process. Conservatively, it would take at least 20 seconds to overheat a worst case capsule if heat removal was completely stopped. Due to the openness of the duct structure, it would be unlikely that a gas bubble could surround a capsule for that long. Thus, no significant gas blanketing of the outside of a capsule is expected.

Internally, if bursting specimens forced the sodium out of the capsule or a passing external bubble allowed the capsule to drain, tests indicate that the capsule will refill with sodium in less than 7 seconds. If 7 seconds are allowed to drain the capsule and another 7 for refilling, no area in the interior could be gas blanketed for more than 14 seconds. This is less than the allowable 20 seconds and would not create a safety hazard.

8.3.2 Solid or Liquid Debris Effects

Due to the large and numerous passages in the duct and the improbability of significant debris generation, external flow blockage is extremely improbable. The sodium flow through the capsule specimen region accounts for less than 10% of the total heat rejection. The temperature control system can easily adjust for this power increase; therefore, blockage of this flow would not cause any excessive temperatures. Thus, no safety hazard will occur due to debris accumulation.

8.3.3 Contamination of Primary Coolant and Cover Gas

The experiment does not contain any fuel or smearable radioactive materials. The impact of tritium generation and its transport to the reactor cover gas and coolant was discussed in Section 8.1 of this document. An analysis of the effect of release of the specimen materials is in "Effects of Release of Materials in Fusion MOTA Test" (Brehm 1988), Appendix J. The conclusion of the analysis is that the amounts of materials is so small that no harmful effects to the reactor would result.

8.3.4 Effect of Exposure of Test Material to Primary Coolant

All test and structural materials in the experiment are chemically compatible with sodium. The analysis referenced in section 8.3.3 addresses this issue and concludes there is no harmful effect to the reactor, although it may result in the loss of experimental data.

8.3.5 Effect of Pin Spacer Failure

No pins are involved in this experiment. Thus, this section is not applicable.

8.3.6 Effect of Subjecting the Failed Element to Normal Sodium Removal Process

Assembly is not subjected to sodium removal; therefore, this section does not apply.

8.3.7 Propagation of Fuel Element Failure Within the Test Assembly

Since no fuel is present in the experiment, no hazard of this kind is possible.

8.3.8 Rupture of Gas Gap System

Leakage of sodium into the gas gap system will only increase the cooling of the capsules, resulting in loss of temperature control but with no safety consequences.

All gas lines which are in direct communication with the reactor cover gas will be equipped with ASME coded check valves. This will prevent escape of cover gas from the reactor vessel through these lines during normal operation.

8.3.9 Overpressure Tritium Handling System Glovebox

This accident scenario is analyzed in "Radiological Assessment of Ex-Vessel Fusion MOTA Accidents" (Prevo 1988), Appendix J. Design features that protect against glovebox overpressure are:

- An automatic pressure control system to maintain pressure in the glovebox in the range of -0.5 to -0.20 inches of water relative to containment.
- Backup pressure control system that will include a pump. The pump, exhausting to CAPS, would be activated if the pressure in the glovebox fell outside the range (e.g., +0.5 to -3.5 inches of water) established for the primary pressure control system.
- The backup pressure control system would also isolate the argon supply if necessary.
- A bubbler type pressure relief system will relieve pressure if either of the two automatic controls fail.

The conclusion of the analysis is that radiological consequences from overpressure of the Tritium Handling System glovebox are acceptable.

8.3.10 Saturate Tritium Recovery System Getter

This accident scenario is analyzed in "Radiological Assessment of Ex-Vessel Fusion MOTA Accidents" (Prevo 1988), Appendix J. This accident also requires the glovebox to be breached, as well as the system piping in order to release tritium to containment.

Design features that protect against saturation of this getter, subsequent to release of tritium to containment and resultant personnel dose, are:

- Tritium getter canisters will be exchanged periodically to avoid saturation.
- Personnel will evacuate in response to a tritium continuous air monitor (CAM) alarm.
- Getter canisters exchanged periodically.
- Personnel could stop sweep gas flow through the canister in response to effluent monitor alarm.

- Tritium break-through is gradual and would be detected by the effluent monitor long before the release rate equalled the production rate.
- Tritium escaping the getter would normally be released to the ventilation exhaust. Release to containment would require another failure such as breach in the glovebox boundary.

The conclusion of the analysis is that radiological consequences from the saturation of the Tritium Recovery System getter are acceptable.

8.3.11 Overheat Tritium Recovery System Getter

This accident scenario is analyzed in "Radiological Assessment of Ex-Vessel Fusion MOTA Accidents" (Prevo 1988), Appendix J. This accident also requires the glovebox to be breached, as well as the system piping in order to release tritium to containment. Features that protect against overheating the getter and subsequent release of tritium are:

- Overtemperature protection includes a dual set point alarm.
 - The first alarm alerts the operator that the temperature has exceeded the normal operating range at a temperature which is below the point at which tritium is released from the getter material.
 - The second alarm activates at a higher temperature and if reached, causes power to the controller to be shut off. The temperature at which it activates is also below the point at which tritium is released from the getter material.
- If the control system failed and caused the getter material to heat up, the rate of temperature increase would be slow due to the limited power capability of the controller. This would cause the tritium to be released gradually over a several-hour period rather than instantaneously.
- If the getter temperature control system failed, any released tritium would be detected by the effluent radiation monitor. Operators would take action to restore the temperature of the getter to within normal operating range in response to an effluent monitor alarm.
- If the getter temperature control system failed, any released tritium would be removed by the downstream backup getter.

The conclusion of the analysis is that radiological consequences from overheating of the Tritium Recovery System getter are acceptable.

8.3.12 Breach Ex-Vessel Sweep Gas Lines

The following features protect against shear of the sweep gas lines:

- The sweep lines will be protected from physical damage by a steel barrier, e.g., channel iron.
- The sweep lines will be routed in or on the floor, thus decreasing the probability of being damaged by moving equipment.
- The sweep gas lines, up to the glovebox, are an extension of the reactor pressure boundary and are ASME, Section III, Class I.
- Operators could close isolation valves and stop sweep gas flow in response to the effluent radiation alarm which would stop the release of tritium.

The conclusion of the analysis is that radiological consequences from breach of the ex-vessel sweep gas lines is acceptable.

8.3.13 Damage to Tritium Handling System Glovebox

This accident scenario is analyzed in "Radiological Assessment of Ex-Vessel Fusion MOTA Accidents" (Prevo 1988), Appendix J. Features that protect against mechanical damage are:

- The glovebox is made of stainless steel with damage-resistant safety glass.
- The glovebox will be secured in place to prevent movement or tipping as a result of normal impacts.
- Operators will take corrective actions in accordance with approved procedure to stop the release in response to effluent radiation monitor alarms.

The conclusion of the analysis is that radiological consequences from damage to the Tritium Handling System glovebox are acceptable.

8.3.14 Loss of Power to Tritium Handling System

This accident scenario is analyzed in "Radiological Assessment of Ex-Vessel Fusion MOTA Accidents" (Prevo 1988), Appendix J. In addition to loss of power, the glovebox and system piping must be breached in order to release tritium to containment. Features that protect against release of tritium to the environment following loss of power are:

- The tritium getter is provided with emergency power.
- Exhaust from the glovebox is monitored for tritium prior to discharge to CAPS.
- Effluent from the combined exhaust which receives CAPS discharge is monitored for tritium prior to discharge to the environment.
- Both the glovebox and effluent exhaust monitors provide local and control room alarms. Operators take corrective actions in accordance with approved alarm response procedures to stop a release.
- Emergency backup power will be provided to the Fusion MOTA experiment.

The conclusion of the analysis is that radiological consequences from loss of electrical power to the Tritium Handling System are acceptable.

8.3.15 Breach Ex-Vessel and In-Vessel BEATRIX-II Sweep Gas Lines

This accident is analyzed in "Radiological Assessment of Ex-Vessel Fusion MOTA Accidents" (Prevo 1988), Appendix J. The source from this accident is the same as the worst source which could result from an HCDA from the BEATRIX-II sweep gas lines. Features that protect against shear of the sweep gas lines are given in subsection 8.3.12. Release of reactor cover gas into containment via the sweep gas lines requires two failures. In addition to shearing the ex-vessel sweep gas lines, failure of the sweep gas lines in-vessel above the sodium level is also required to release reactor cover gas into containment. The configuration of the test assembly between the top of the sodium pool and reactor head is such that failure in this area is highly unlikely.

The conclusion of the analysis is that radiological consequences from simultaneously breach of the ex-vessel and in-vessel sweep gas lines are acceptable.

8.3.16 Breach Nonvented Capsule in Interim Examination and Maintenance Cell

This accident is analyzed in "Radiological Assessment of Ex-Vessel Fusion MOTA Accidents" (Prevo 1988), Appendix J. Features that protect against release of tritium to the IEM cell and then to the environment are:

- The lithium oxide sample material would retain most of the tritium and thus limit release.
- The tritium is encapsulated in stainless steel.

The conclusion of this analysis is that radiological consequences from breach of a nonvented capsule in the IEM cell are acceptable.

8.3.17 Expulsion of Sodium from Reactor Pressure Vessel through BEATRIX-II Sweep Gas Lines

Sodium that may be expelled from the BEATRIX-II 0.054" inside diameter sweep gas tubes following failure of the tubes both below the sodium level and ex-vessel would be insignificant compared to the 240 pounds of sodium predicted to be expelled to the Center Island from the Spool Piece seals as a result of the HCDA (FS-257 1976).

There is no credible failure (other than an HCDA) of either the solid pellet or temperature change capsules which could cause sodium to migrate above the reactor head. Even if it could reach a point above the reactor head, it would quickly freeze and the volume in the 0.054 inch inside diameter tubes would be small and contained.

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