SIRIUS-M: A Symmetric Illumination, Inertially Confined Direct Drive Materials Test Facility

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ABSTRACT

SIRIUS-M is a fusion materials test facility designed to duplicate the time-dependent radiation damage structure unique to ICF systems in order to provide the technology base necessary for an ICF demonstration facility.

Single-shell ICF targets are symmetrically illuminated by 32 beams of a KrF laser with a total laser energy of 1 MJ. A wall loading of 2 MW/m^2 is achieved at a repetition rate of 10 Hz and target gain of 13.4. Xenon gas at a pressure of 133 Pa (1 torr) is placed in the 2 m radius, graphite-tiled cavity in order to protect the first wall from the x-rays and ions produced by the explosions.

Two circular test modules are used in SIRIUS-M. Each module has a front surface area of 1 m^2 and fits between three beam ports. No significant radial and azimuthal damage variation in the module results from these penetrations. The peak dpa rate is 24 dpa/FPY yielding a peak accumulated damage of 120 dpa at the end of life of the SIRIUS-M facility. A total volume-integrated-damage figure of merit of 2,840 dpa-ℓ per full power year can be achieved in SIRIUS-M.

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1. Introduction

The need to test structural materials under realistic fusion reactor conditions has been discussed in both the magnetic confinement fusion (MCF) and inertial confinement fusion (ICF) communities for over a decade. Irradiating small size materials samples in a neutron flux can be accomplished in fission reactors or small DT neutron source facilities. However, the restricted temperature range and small individual test volumes, along with serious neutron energy spectral differences, make complete testing of materials in these facilities impossible. The MCF program has taken the lead in attempting to solve this problem by sponsoring several test reactor studies such as FERF [1], TETR [2], INTOR [3], TASKA [4], TASKA-M [5], TDF [6], and FEF [7]. Most of these studies have concentrated on providing a nuclear and thermal environment which would closely simulate that to be expected in the first demonstration reactor or the first commercial magnetic fusion reactor.

In contrast to the MCF technology program, the efforts of the ICF technology program have been focused on conceptual design of commercial power plants; there has been a curious lack of near term test facility designs. The singular exception is a brief scoping study of a device called LA FERF [8] in 1975 at the Lawrence Livermore National Laboratory. It is commonly assumed by the ICF community that the MCF materials program will provide the data needed for designing inertial confinement reactors. However, the large differences between the damage conditions in ICF and MCF environments arising from geometrical, spectral, and temporal effects, as quantified by Kulcinski and Sawan [9], make it necessary to develop a dedicated ICF materials test facility. To this end, the Fusion Technology Institute of the University of Wisconsin (FTI) and the University of Rochester's Laboratory for Laser Energetics (LLE), in cooperation with the Naval Research Laboratory (NRL), have conducted a study of the critical issues related to the design of an ICF materials test facility, SIRIUS-M [10]. The facility is designed to duplicate the time-dependent radiation damage structure unique to ICF systems, in order to provide the technology base necessary for an ICF demonstration facility.

In selecting the technical specifications and design parameters for the SIRIUS-M facility, we have attempted to limit the initial capital and operating costs by limiting the "mission" of the facility to only materials testing; tritium breeding and high-temperature recovery of thermonuclear energy are not included. The "desired" neutron wall loading has been set at $\gtrsim 2$ MW/m^2. This corresponds to an irradiation of ~ 1 $MW\text{-}yr/m^2$ per calendar year of operation so that the necessary cumulative damage can be achieved in a few years of operation. Attention has been focused on several areas unique to an ICF materials test facility including: test module design and damage rate estimation, cavity design and first wall protection, target design, and placement of the final mirrors.

This paper provides an overall description of the SIRIUS-M facility, along with detailed analyses of cavity design and test module performance. Additional details may be found in two companion articles [11,12].

2. Design Parameters and Facility Design

Table 1 lists the main design parameters for the SIRIUS-M facility. Single-shell ICF targets are symmetrically illuminated by 32 beams of a KrF laser equidistantly distributed around a 2 m radius spherical cavity. The total laser energy is 1 MJ. The beam arrangement is based on a twenty sided icosahedron, where the sides are equilateral triangles superimposed on a spherical surface. The 32 beams penetrate the cavity at the centers of the twenty triangles and the twelve vertices where they meet.

Table 1. Design Parameters for SIRIUS-M

	Value
Fusion power	134 MW
Tritium consumption rate	3.7 kg/CY
Target yield	13.4 MJ
Target gain	13.4
Repetition rate	10 Hz
Laser energy (KrF)	1 MJ
Number of laser beams	32
Neutron wall loading	2 MW/m^2
Chamber inner radius	2 m
Cavity gas	xenon
Gas pressure	1 torr
Xenon inventory	1600 liters (STP)
Tritium inventory in Xe processor	114 g
Number of tiles	20
Tile area	2.5 m^2/tile
Face material	graphite
Tile thickness	1.0 cm
Back material	HT-9
Coolant	water
Module diameter	1.14 m
Module depth	0.2 m
Capsule diameter	5 cm
Capsule length	20 cm
Capsule volume	0.39 liters
Number of capsules	434
Active test volume	171 liters
Maximum dpa/FPY (Fe)	24 dpa/FPY
Maximum appm He/FPY (Fe)	145 appm/FPY

Reasonably achievable values of target gain, 13.4, and repetition rate, 10 Hz, are assumed so that a total fusion power of 134 MW is obtained. Detailed design and performance of single-shell ICF targets may be found in Ref. [13]. In order to achieve the desired neutron wall loading of 2 MW/m^2, a cavity radius of 2 m is used. The cavity is surrounded by actively-cooled, graphite-faced tiles followed by a 40 cm thick lead reflector, a 30 cm thick steel reflector, and a biological shield (see Fig. 1).

The first wall consists of twenty water-cooled graphite faced tiles shaped as equilateral triangles, 2 m on edge (see Fig. 2). The tiles consist of a 2 cm thick ferritic stainless steel (HT-9) base structure with cooling channels machined in it. The base structure has a collar in the center which is the primary support for the tile. Cooling line fittings are built into the support collar. A 1 cm thick graphite surface is brazed to the front of the tile. The graphite also has a collar which extends into the base structure collar. The central collar of each tile serves as a beam port while each of the vertices subtend one fifth the circumference of a beam port.

Each tile is supported only on the central collar. The tiles are unrestrained at any other point and in this sense will not be subjected to high thermal stresses. Four tiles are modified to accommodate the materials test modules. Two such modules are used and are located diametrically opposite to one another. Each test module fits between three beam ports. It has a front surface area of 1 m^2 which represents 2% of the solid angle seen by the target. Additional details of test module design and performance are given in a later section.

In order to achieve a neutron wall loading of 2 MW/m^2 at a reasonable repetition rate (10 Hz) and target yield (13.4 MJ), it is necessary to protect the graphite tiles by placing low pressure xenon gas (133 Pa) in the cavity.

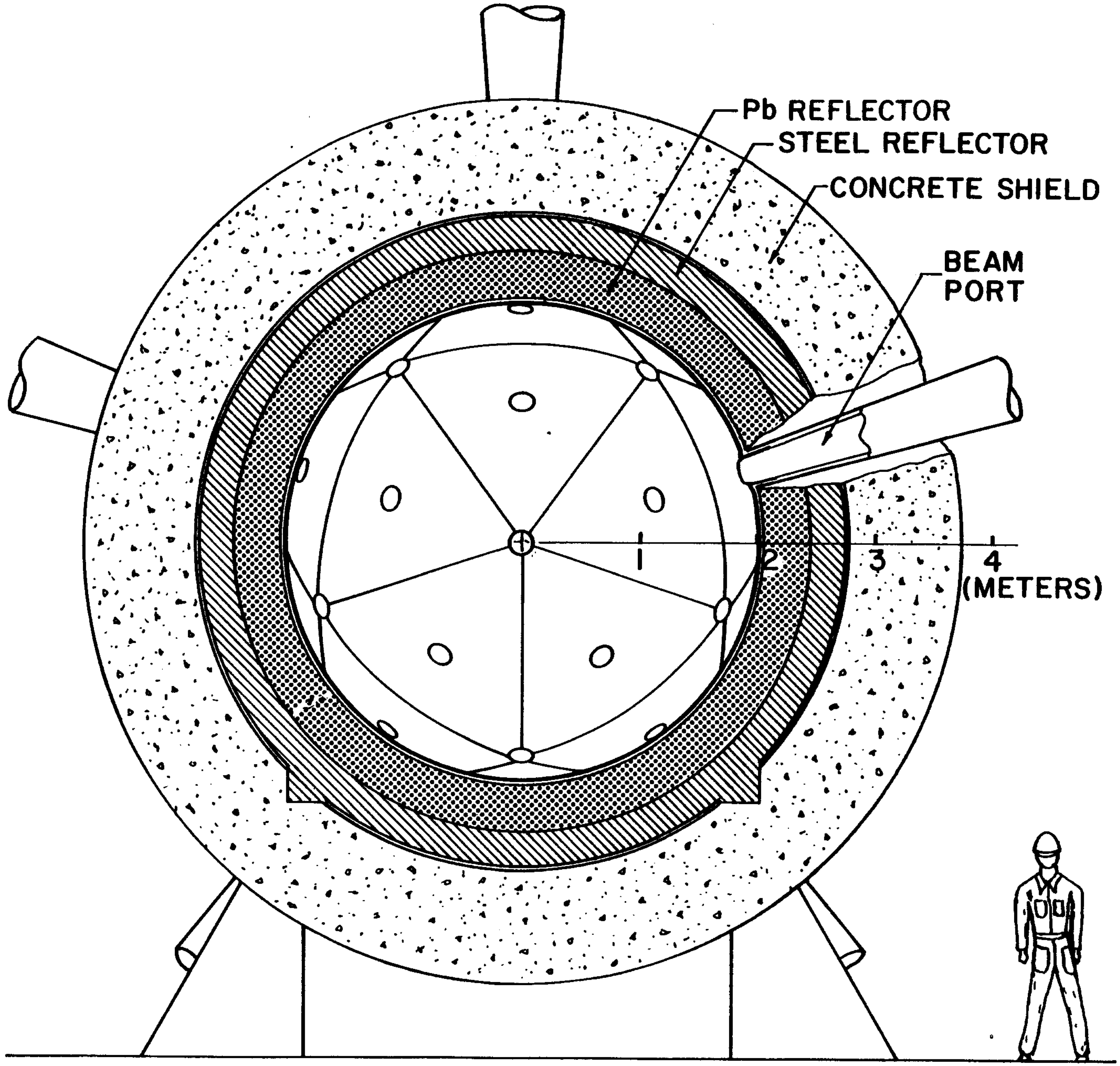

Fig. 1. SIRIUS-M cavity, reflector and shield.

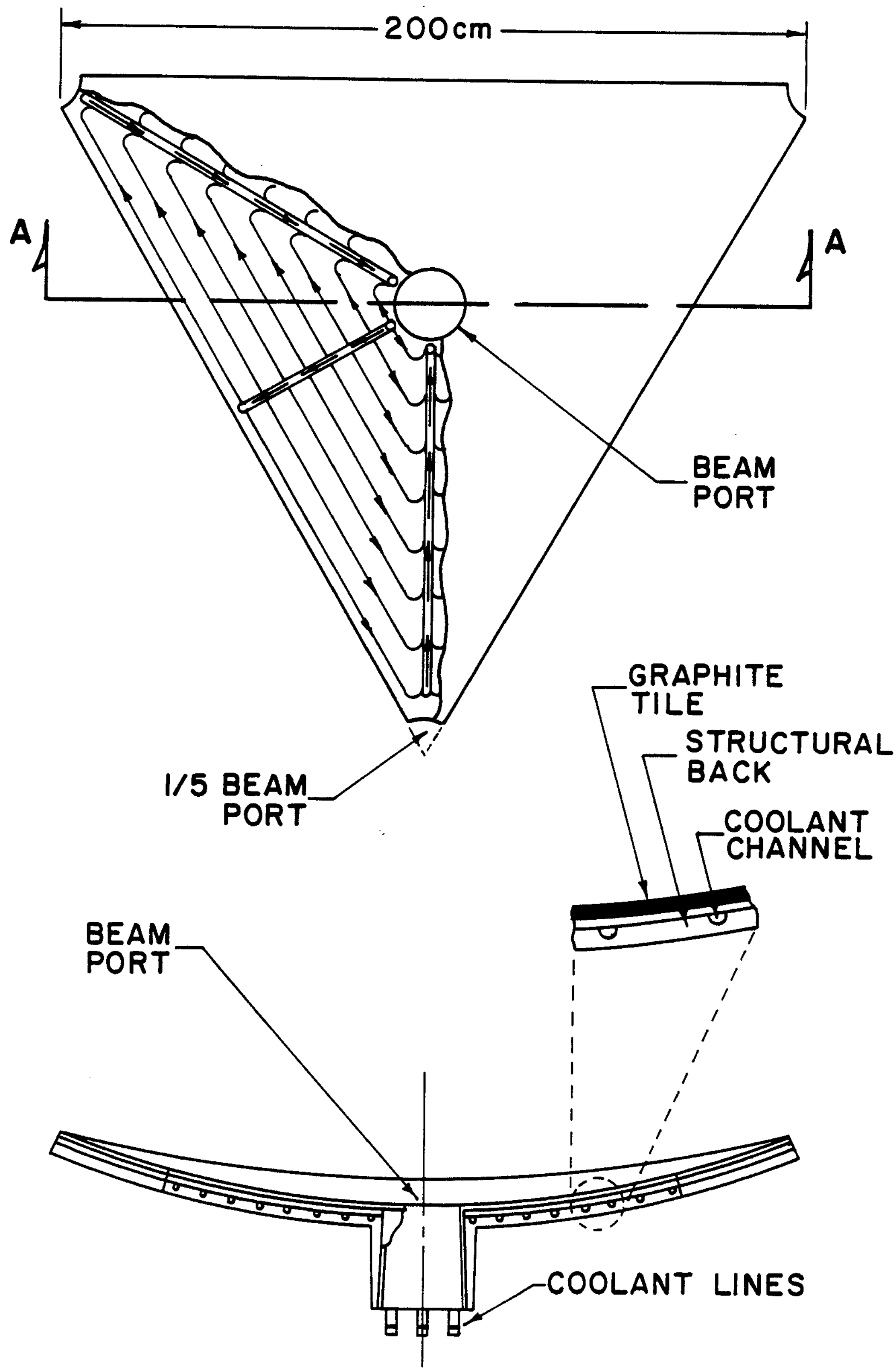

Fig. 2. Actively cooled graphite-faced tile.

Gas protection is based on the principle that the soft x-rays and ionic debris produced by the explosions will be stopped in the gas which reradiates that energy to the wall over a relatively "long" period of time ($\sim 10^{-4}$ s) and thus limits the wall surface temperature rise, evaporation, and compressive stress. The use of gas protection in SIRIUS-M with a wall loading of 2 MW/m^2 results in negligible surface evaporation and acceptable maximum thermal stress (see next section).

A gas handling system has been designed in order to process the exhaust gases from the reactor chamber, prepare decontaminated xenon gas for reinsertion into the reactor cavity, and recover, purify, and isotopically-separate DT for the preparation of new target fuel. This processing must be accomplished with low loss and minimal inventories of xenon and tritium because of the high cost of Xe ($10.20/liter (STP)) and tritium ($10,000/g) and the radiological hazard caused by tritium escaping to inhabited areas.

The xenon residence time in the cavity is limited to 1 s in order to maintain the level of impurities in the chamber gas at an acceptable level for beam propagation and focusing. The reactor cavity exhaust consists of xenon, unburned fuel, He ash, target shell debris $(CH_2)_x$, and some methane and acetylene formed by reaction of the hydrogen isotopes with the hot graphite first wall. Detailed design of the gas handling and unburned tritium recovery system may be found in Ref. [10]. The inventories of xenon and tritium in various processing equipment are modest, totaling about 1660 liters (STP) and 114 g, respectively.

It should be noted that since no tritium is bred in SIRIUS-M, the tritium demand will have to be purchased from government-operated production facilities. For a capacity factor of 50%, the tritium consumption rate will be 3.7 kg/CY. While this amount is less than the production rate of one Savannah

River type facility, it is clear that tritium costs will represent a significant (but affordable) fraction of the operational costs of SIRIUS-M. A comparison between the tritium demand for various test facilities is given in Table 2.

Figure 1 shows the Pb reflector as a continuous spherical shell, 40 cm thick which is penetrated by the 32 beam ports. The reflector will have two additional penetrations for accommodating the material test modules. As presently envisaged it consists of two concentric spherical HT-9 shells, with the inner shell supported on the outer by means of the beam tubes. The space between the shells is filled with molten lead which can be circulated for cooling, or separately cooled by tubes embedded in it. Pressurized water cooling (as in the case of the tiles), steam, or helium gas can be utilized. Since 40 cm of lead is a very heavy load, the primary support structure for the reflector will be the 30 cm thick steel reflector which is made of HT-9. In principle, there is no reason why the outer shell of the reflector cannot support itself. This will depend on where the vacuum boundary is located and on the cooling connections. Fabrication considerations may also preclude the possibility of combining the Pb reflector and steel reflector into a single structure. These decisions can only be made when a more detailed design is available.

During operation, the beam tubes (and cavity) will be filled with xenon gas up to a point where some kind of window will separate it from the environment of the laser. Just where the vacuum boundary is located is very critical in the design of the cavity, since this boundary will experience one atmosphere of pressure. If the reflector is used as the vacuum boundary, the final mirrors will have to be sealed within, and supported on, the beam tubes. The reflector and beam tubes will then be the vacuum boundary for the cavity.

Table 2. Performance and Design Parameters of Various Facilities

Facility	Type	Dpa-ℓ	Fusion Power (MW)	Ave. Wall Loading (MW/m^2)	TBR (-)	T_2 Purchased (kg/FPY)
RTNS-II	Accelerator	0.0003	---	---	---	~ 0
FMIT	Accelerator	5	---	---	---	~ 0
INTOR	MCF-Tokamak	182	620	1.3	0.6	13.8
TASKA-M	MCF-Mirror	530	6.8	1.0	0	0.4
TASKA	MCF-Mirror	1510	86	1.5	> 1	0
FEF	MCF-Mirror	970	~ 12	2.9	0	0.7
SIRIUS-M	ICF-Laser	2840	134	2.0	0	7.4

Another option is to put the vacuum boundary further back, beyond the final focusing mirrors. This would obviate the need for beam tubes between the cavity and the final mirrors, and the laser light would simply be focused into the holes through the cavity. An obvious advantage here is that the cavity can be isolated from the mirror support structure and mirror maintenance becomes easier. The disadvantage is that the vacuum chamber is much larger and more xenon gas will be needed.

Pumping requirements at the 133 Pa (1 torr) cavity pressure are minimal and can be adapted to either of the two options discussed above. Several of the beam tubes can have pumping ports attached to them in the case of the first option. In the second option a pump station will be located at the vacuum boundary and the gas simply exhausted through the beam port holes in the cavity.

Routine maintenance will be required for the final mirrors, graphite tiles, and coolant connections in the back of the shield. The final mirrors are directly exposed to neutrons, and will have to be replaced on a yet-to-be-determined schedule. Support structures in the vicinity of the final mirrors will be activated and, therefore, this function will most likely require remote maintenance. Further, access to the final mirror will be impeded by beam tubes and other structures in the way. Replacement of the mirrors will have to be accomplished by a special purpose maintenance machine which can travel on guides to each mirror location. Final alignment of the mirror can be accomplished remotely from the control room.

Graphite faced tiles may need replacing sometime during the life of the reactor. Access to the inside of the reactor cavity will be needed for this purpose. The 1.14 m diameter penetrations for the material test modules can be used for insertion of a remote control special purpose device which can service the tiles. Some modification of one of the penetrations may have to be made to be able to insert a tile which is ~ 2 m at its widest point.

Finally, some provision must be made for servicing the coolant lines at the back of the cavity. A special purpose machine designed to maneuver in the space behind the shield will be needed for this task. General purpose manipulators mounted on the machine should be able to fulfill the requirements of maintaining the coolant lines. Limited "hands on" maintenance will make this function much easier.

3. First Wall Design Considerations

The thermal response of the SIRIUS-M graphite-tiled first wall (FW) due to the different types of radiation released during the microexplosion of the target has been analyzed for both unprotected and gas-protected walls. The corresponding evaporation rate and thermal stresses have also been calculated. Based on these analyses, it has been concluded that the maximum allowable wall loading for an unprotected cavity will be considerably lower than the required design value of 2 MW/m^2. Such a wall loading can be successfully accommodated by using gas protection. This section briefly describes the methodology and results obtained in the first wall design analysis. Additional details may be found in Refs. [10,12].

As indicated earlier, the targets used in SIRIUS-M have a yield of 13.4 MJ, of which 0.8 MJ are carried by the x-rays, 2.6 MJ are carried by the ions, and the remainder by the neutrons. The x-rays, ions, and neutron spectra for the SIRIUS-M target are shown in Fig. 3. The x-rays are assumed to be emitted in ~ 20 ps. In addition to x-rays, ions, and neutrons, the FW is irradiated by the portion of the laser light reflected from the target. The KrF laser (λ = 0.248 μ) used in SIRIUS-M uniformly delivers 1 MJ to the target, in about 11 ns. It is assumed that 10% (i.e., 100 KJ) of the laser energy is reflected

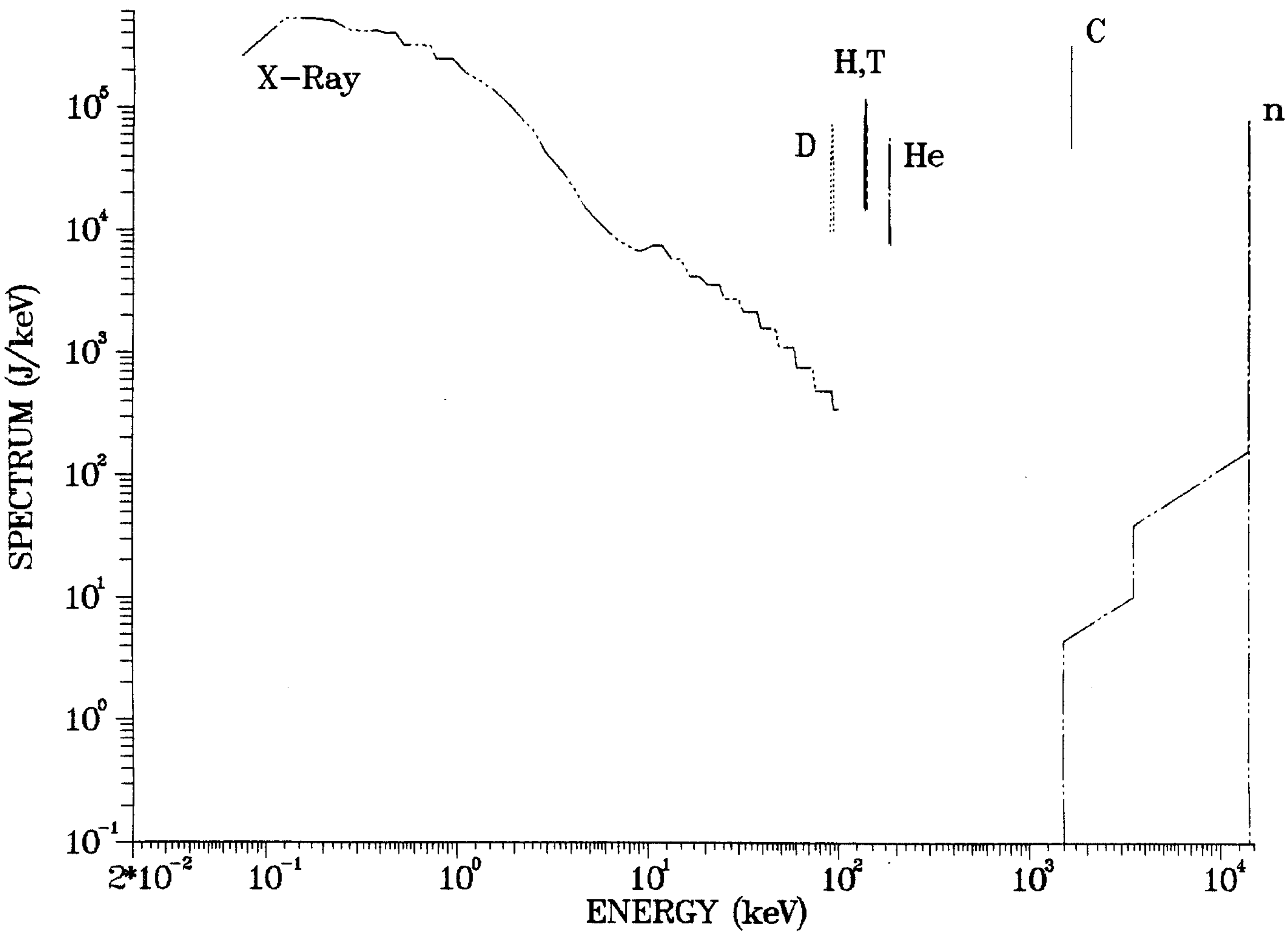

Fig. 3. X-ray, ion and neutron spectra for the SIRIUS-M target.

and refracted from the target [14] and is ultimately absorbed by the first wall. Depending on the reflectivity of the FW materials, part of this laser energy will be absorbed while the remainder will be reflected. In the spherical geometry of SIRIUS-M, and with the assumption of uniform and normal incidence of the light on the wall, the reflected portion from one location on the wall will strike the opposite side. In effect, a train of laser pulses with rapidly attenuated amplitudes will strike the FW. A monochromatic hemispherical emissivity of 50% has been assumed for the graphite tiles at 0.248 μ.

A parametric study has been conducted to determine the response of unprotected graphite tiles to the various components described above. The transient temperature history within the first wall has been calculated for different radii and different steady state temperatures. The steady state temperature is the front surface temperature reached before the following microexplosion debris reaches the wall. The value of the steady state temperature depends on the method used to cool the first wall. For the actively cooled tiles used in SIRIUS-M, the steady state surface temperature is ~ 500°K. Modified versions of the T*DAMEN [15] and A*THERMAL [16] computer codes which account for temperature dependence of the wall physical properties have been used in these calculations.

Figure 4 shows the temperature rise at the front surface of the wall for different values of the cavity radius, as a function of time for a steady state temperature of 500°K. These results show that the surface temperature reaches its steady state value (i.e., temperature rise = zero) in approximately 1 ms, well before the following explosion which occurs 100 ms later (rep. rate = 10 Hz). The evaporation rates for the different temperature histories shown in Fig. 4 have been calculated. For an evaporation limit of 1 mm/FPY, the minimum cavity radius would be 3.25 m for T_b = 500°K. This radius corre-

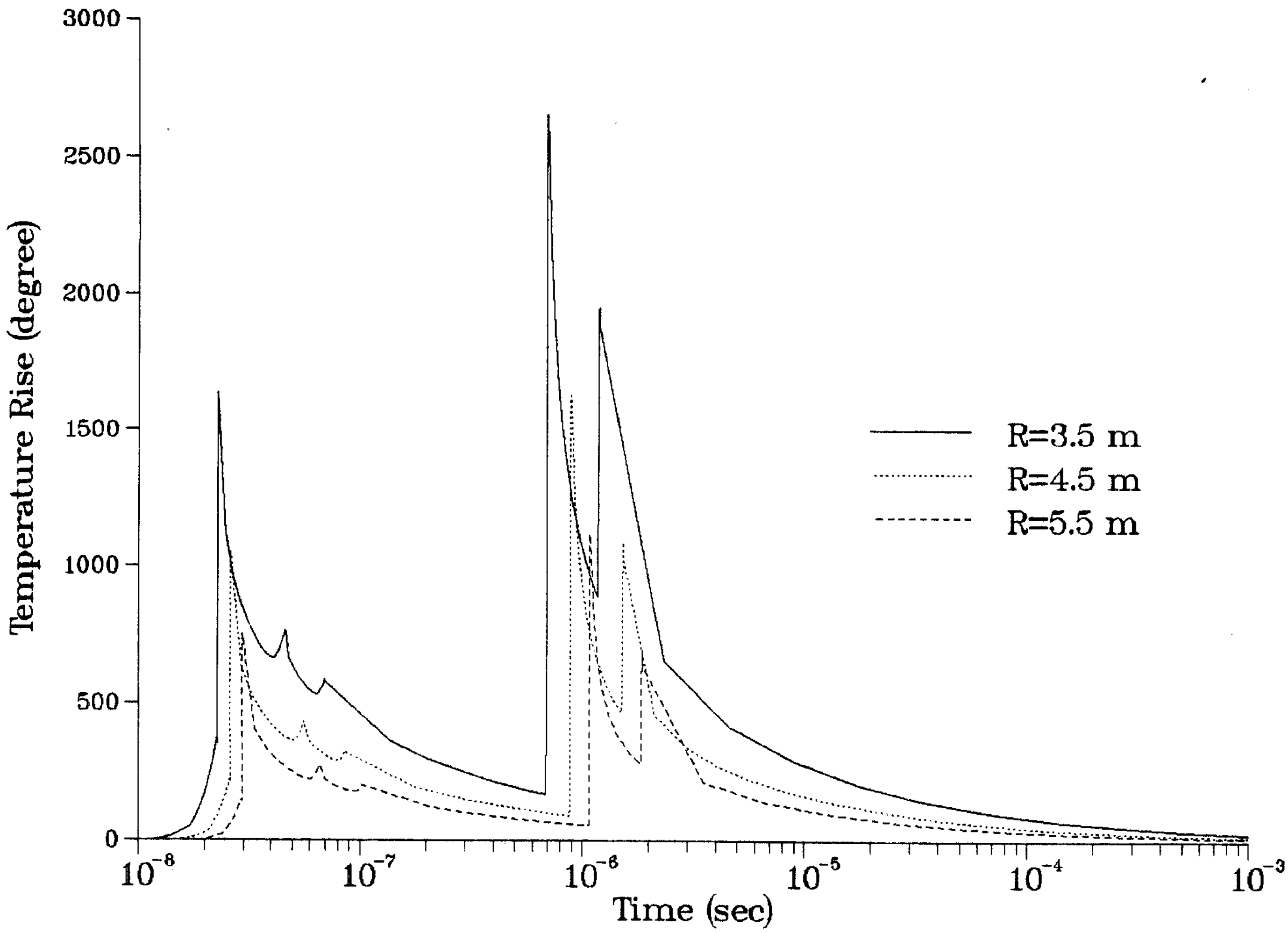

Fig. 4. Temperature rise at the front surfaces of the FW for a cavity radius of 3.5, 4.5 and 5.5 m, with a steady state temperature of 500°K.

sponds to a 0.76 MW/m^2 neutron wall loading which is considerably less than the required goal of 2 MW/m^2 for SIRIUS-M (see Fig. 5). In addition, calculations have been made to determine the thermal stresses in the first wall due to the temperature histories shown in Fig. 4. By setting the peak compressive stress at the front surface equal to the compressive strength of the graphite (at the surface temperature), the minimum cavity radius is obtained. This limit is labeled as the "stress limit" in Fig. 5. These results show that from a thermal stress standpoint the maximum allowable wall loading for unprotected graphite walls is limited to ~ 0.3 MW/m^2 which is well below the design goal of 2.0 MW/m^2. Hence, gas protection is used.

Gas protection is based on the principle that the soft x-rays and ionic debris produced by the explosions will be stopped in the gas which reradiates that energy to the wall over a relatively long period of time (~ 10^{-4} s) and thus limits the wall surface temperature rise, evaporation, and compressive stress. The MFFIRE [17] code used to analyze this problem is a Lagrangian hydrodynamics multigroup radiative heat transfer finite difference computer code. It is assumed that the x-ray and ion energies are deposited in the gas over a short time compared to the time scales for radiative heat transfer and hydromotion in the gas so that the energy density from deposition is treated as an initial condition. The gas is then allowed to radiate and hydrodynamically move as it will. Radiative heat transfer is modeled within a 20 group flux-limited diffusion approximation, where the group opacities are provided by MIXERG. The code provides the radiative heat flux and the shock pressure on the first wall as functions of time. For a 2 m radius cavity with xenon pressure of 133 Pa (1 torr), the heat flux is shown in Fig. 6. The corresponding maximum pressure is only 0.14 atm which is too low to cause any damage to the first wall tiles.

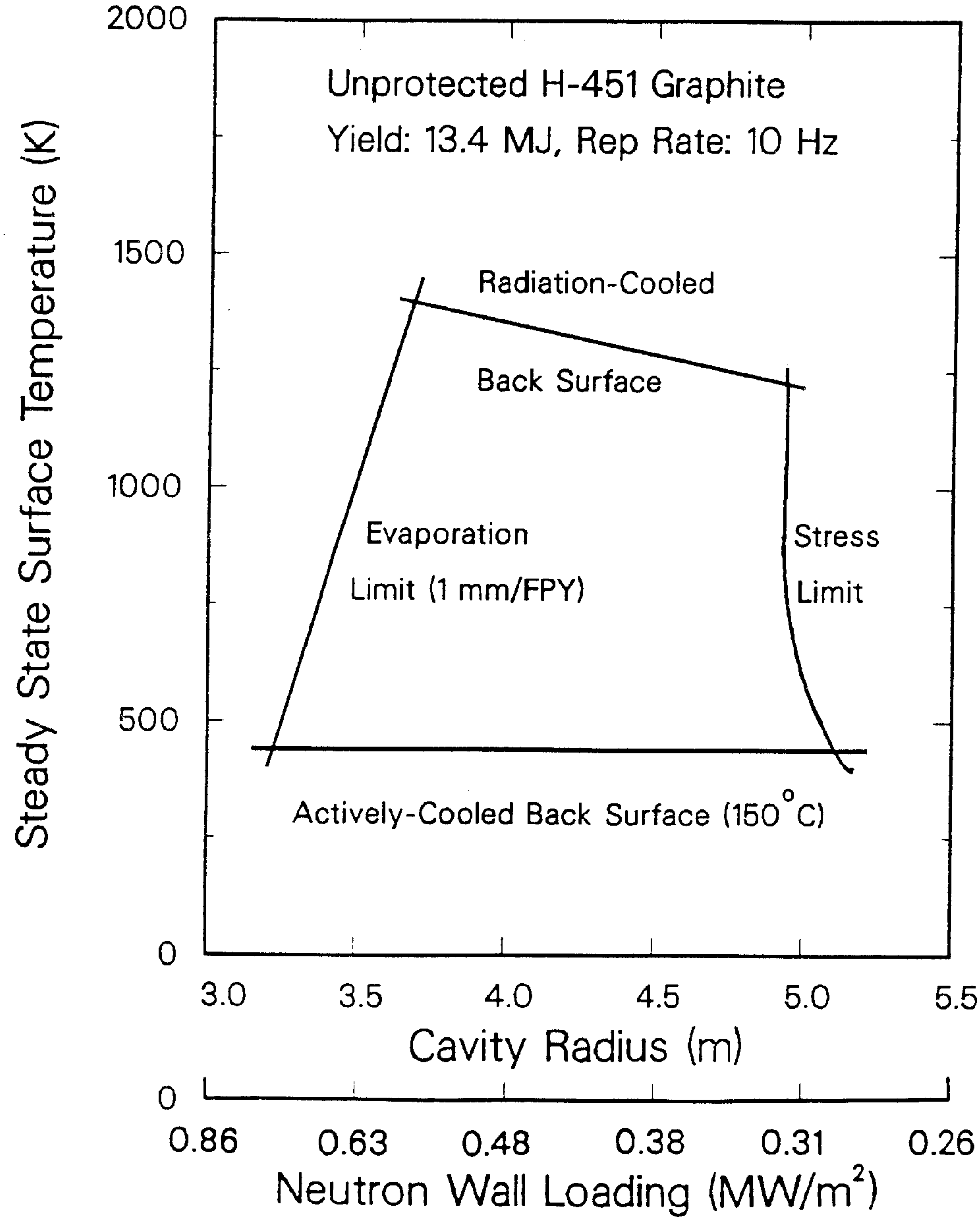

Fig. 5. Evaporation and stress limits for dry wall cavities.

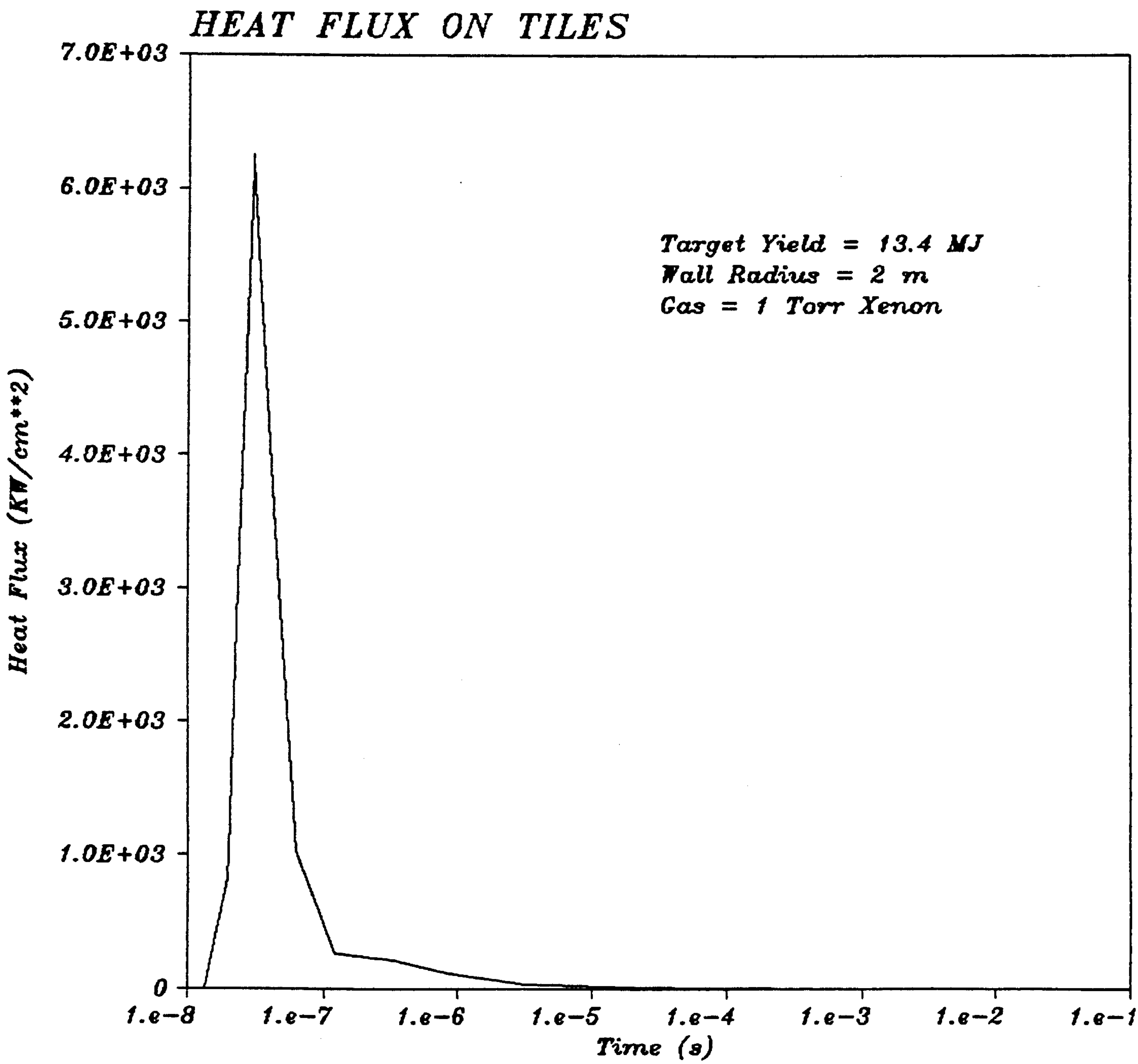

Fig. 6. Radiation heat flux on the first wall of the SIRIUS-M target chamber.

The thermal response of the wall to the radiated heat flux from the gas, along with the unattenuated hard x-rays ($\gtrsim$ 3 keV) and ions, and reflected laser light has been calculated. For a 2 m radius cavity, i.e. a neutron wall loading of 2 MW/m^2, the surface temperature response is shown in Fig. 7. Surprisingly, the design is limited by the temperature rise, i.e. compressive stress, caused by the reflected laser light. The reradiated energy is spread over a long enough time to reduce the surface temperature rise considerably. The reflected laser light is, however, deposited over an extremely short time and is, therefore, more limiting. When the steady state surface temperature is added to these temperature rise values, the maximum surface temperature is 1666°K. Based on this surface temperature history, the surface evaporation rate was found to be negligible. The maximum thermal stress, however, was found to be only 12% lower than the compressive strength of graphite at the peak surface temperature.

4. Materials Test Modules Design and Performance

The test module should be designed such that the test specimens accumulate high damage levels in the shortest possible time. The test module in SIRIUS-M is located at a distance of 2 m from the target resulting in a neutron wall loading of 2 MW/m^2 at the front surface of the module. A 2 mm thick graphite liner is used on the module surface to protect it from the cyclic heat flux. This results in a 2.4% drop in the peak dpa rate achievable in the test specimens. An effort was made to maximize the damage rate by appropriate choice of reflector material and location. A lead reflector surrounding the test module was found to double the achievable damage level in the module compared to a cantilevered test module design where a steel reflector is located 8 m from the target.

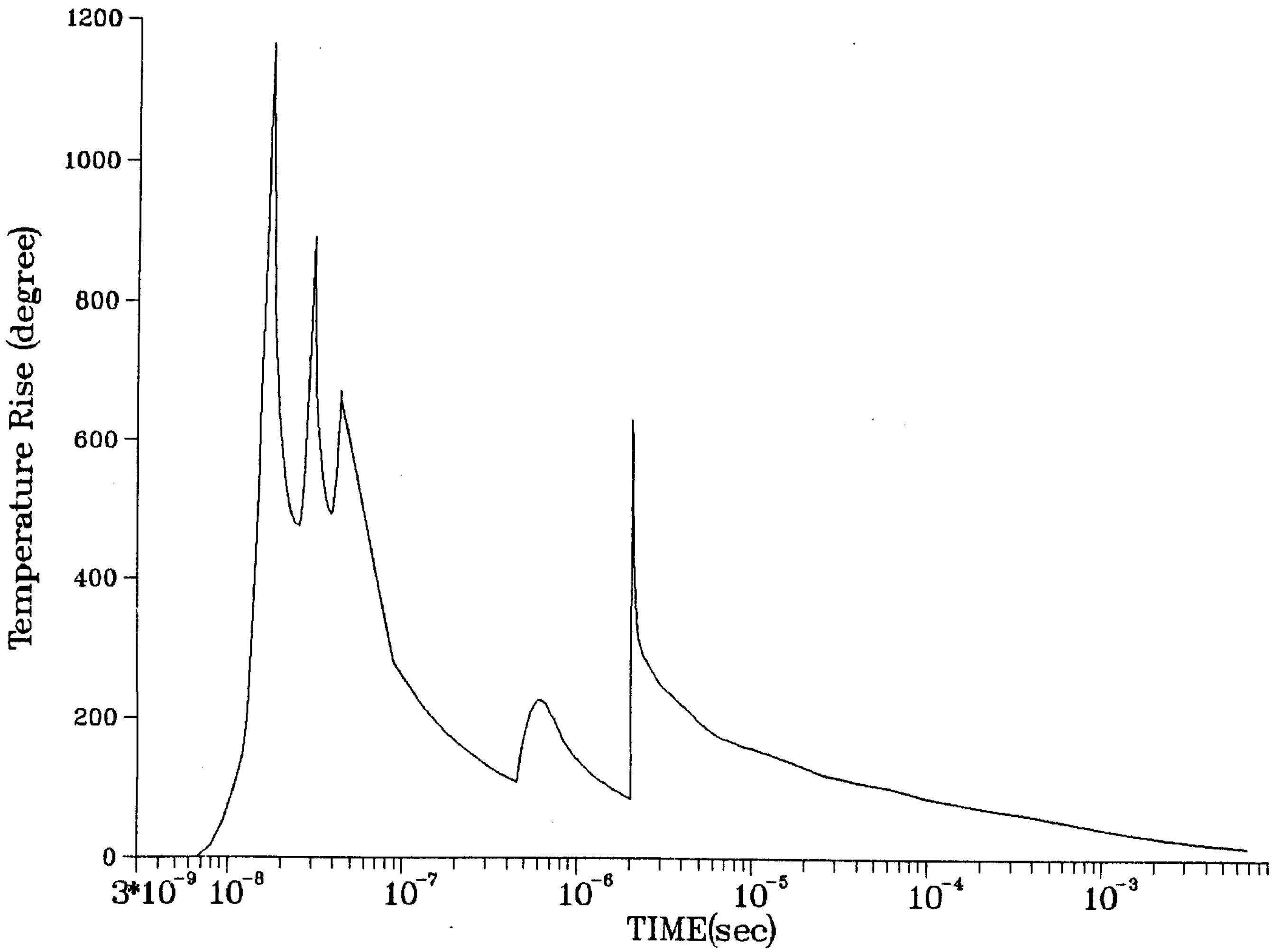

Fig. 7. Temperature rise in graphite tiles in SIRIUS-M for base case.

Two circular test modules are utilized in SIRIUS-M and are located on diametrically opposite sides of the chamber. Each module has a front surface area of 1 m^2 and fits between three beam ports. The test capsules are 5 cm in diameter and 20 cm in length. They are placed perpendicular to the front surface of the module. Each module uses 217 capsules providing 85.4 liters of test volume which represents 40% of the 213.5 ℓ module volume. The capsules consist of 50% NaK, which acts as a thermal contact material, and 50% 316 SS, which represents the specimens and capsule structure. The HT-9 module structure represents 20% of the module volume. The helium gas coolant occupies the remaining volume. The large pressure of the helium coolant (~ 3400 kPa) requires using curved module front surfaces. The test module for SIRIUS-M consists of two parts: a quick access cylindrical part and a long term annular part. Both parts have 3 mm thick HT-9 semi-ellipsoidal pressure heads that protrude 8 cm into the chamber. The quick access cylindrical part is used for short term test specimens. The geometrical configuration of the two parts of the test module is shown in Fig. 8.

One-dimensional coupled neutronics and photonics calculations have been performed to give estimates for the axial variation of power density and damage rate in the test capsules. The helium production rate drops faster than the dpa rate as one moves towards the back of the module. The He/dpa ratio is 5.8 at the front and drops to 3 at the back. The peak and average power densities in the test capsule are 11.2 and 8 W/cm^3, respectively. The power to be removed from each capsule is 3.2 kW and the power generated in each test module is 1.3 MW.

Three-dimensional neutronics calculations have been performed to determine the damage profiles and testing capabilities of the materials test modules. No significant radial or azimuthal variation in damage was observed

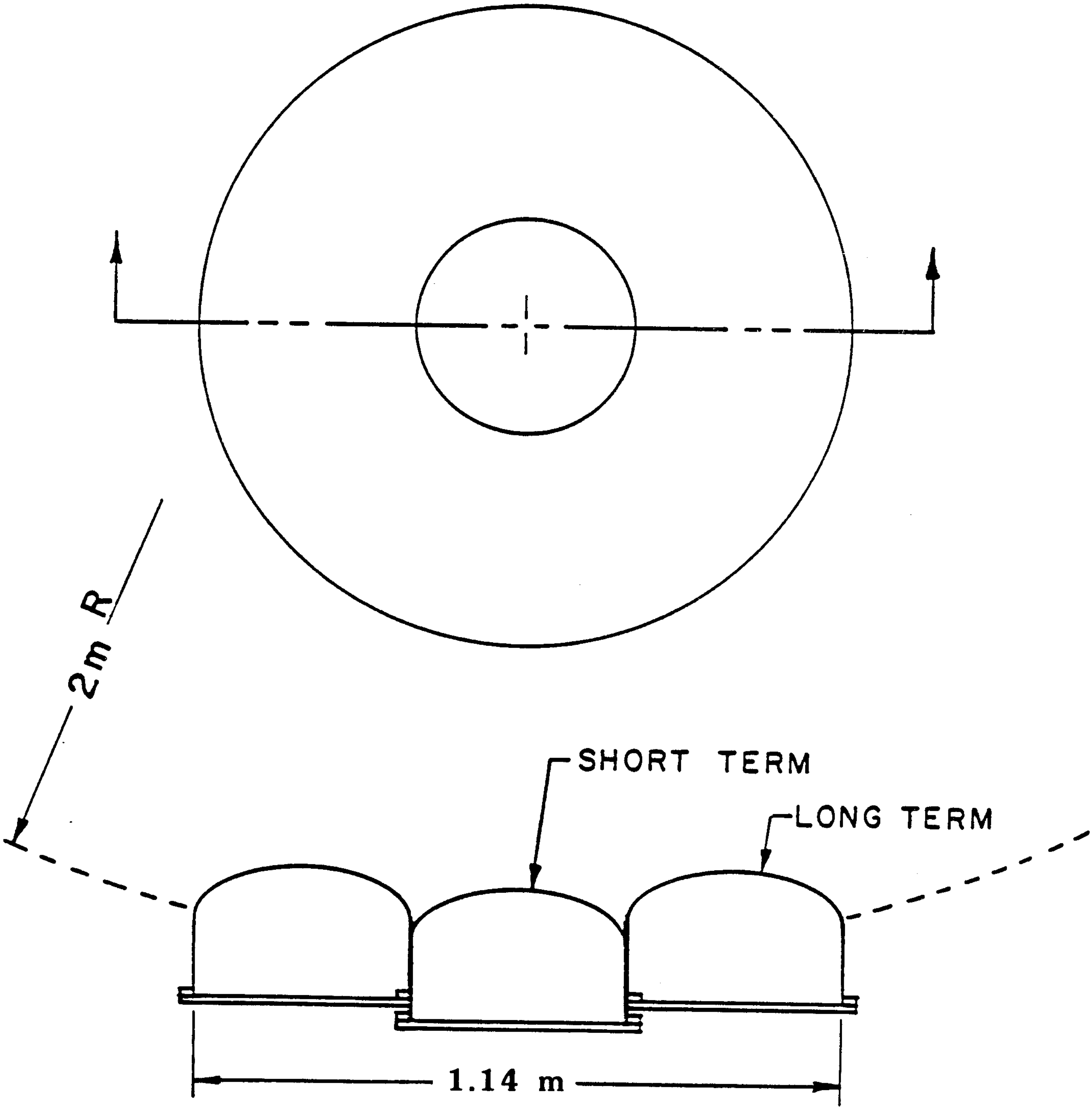

Fig. 8. Geometrical configuration of the test module.

as a result of the three beam ports surrounding the test module. The peak dpa and helium production rates in the module were calculated to be 24 dpa/FPY and 145 He appm/FPY, respectively. The lowest values at the back of the module are 12 dpa/FPY and 36 He appm/FPY. This modest drop, compared to that in MCF test facilities [4,5], is due to the nature of the neutron source (point versus volume) and geometrical differences between ICF and MCF reactors that yield smaller damage gradients in the spherical ICF reactors [9,11]. The small axial variation of damage in SIRIUS-M suggests that longer test capsules can be utilized in ICF test modules.

The three-dimensional results were used to determine the value for the volume integrated damage accumulated at the end of 10 calendar years of operation at 50% availability. The total dpa-ℓ value obtained in the two test modules of SIRIUS-M is 14,200. Table 2 gives a comparison between the values of dpa-ℓ per FPY of operation for SIRIUS-M, the proposed MCF test facilities (INTOR, TASKA and TASKA-M), and the high energy neutron source test facilities (FMIT and RTNS-II). The dpa-ℓ figure of merit in SIRIUS-M is a factor of 1.9 higher than that in TASKA and much higher than the corresponding values in the other test facilities.

5. Conclusions and Recommendations

The conclusions and recommendations of the SIRIUS-M study are the following:

1. Inertial confinement fusion offers the opportunity to build a low power, non-tritium breeding, high performance materials test facility. This might include a 1 MJ short wavelength laser, low gain targets (~ 10), large materials test volume, low damage gradient within the test volume, a small fusion reaction cavity, and efficient geometry for neutron multiplication.

2. The crucial issues that must be faced for the materials test facility are a short wavelength laser with high repetition rate, survivable optics consistent with uniform illumination, uniformity of target irradiance, laser beam focusing through 133 Pa (1 torr) of gas, optics positioning, stability, and layout, and reflected laser light heating of the first wall.
3. It is recommended that those issues amenable to experimental investigation, such as optics damage and laser focusing in the presence of a gas, be investigated. This will greatly improve the level of understanding of supporting technology for inertial confinement fusion.

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