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SOME APPLICATIONS OF FISSION-BASED TESTING CAPABILITIES IN THE DEVELOPMENT OF FUSION TECHNOLOGY

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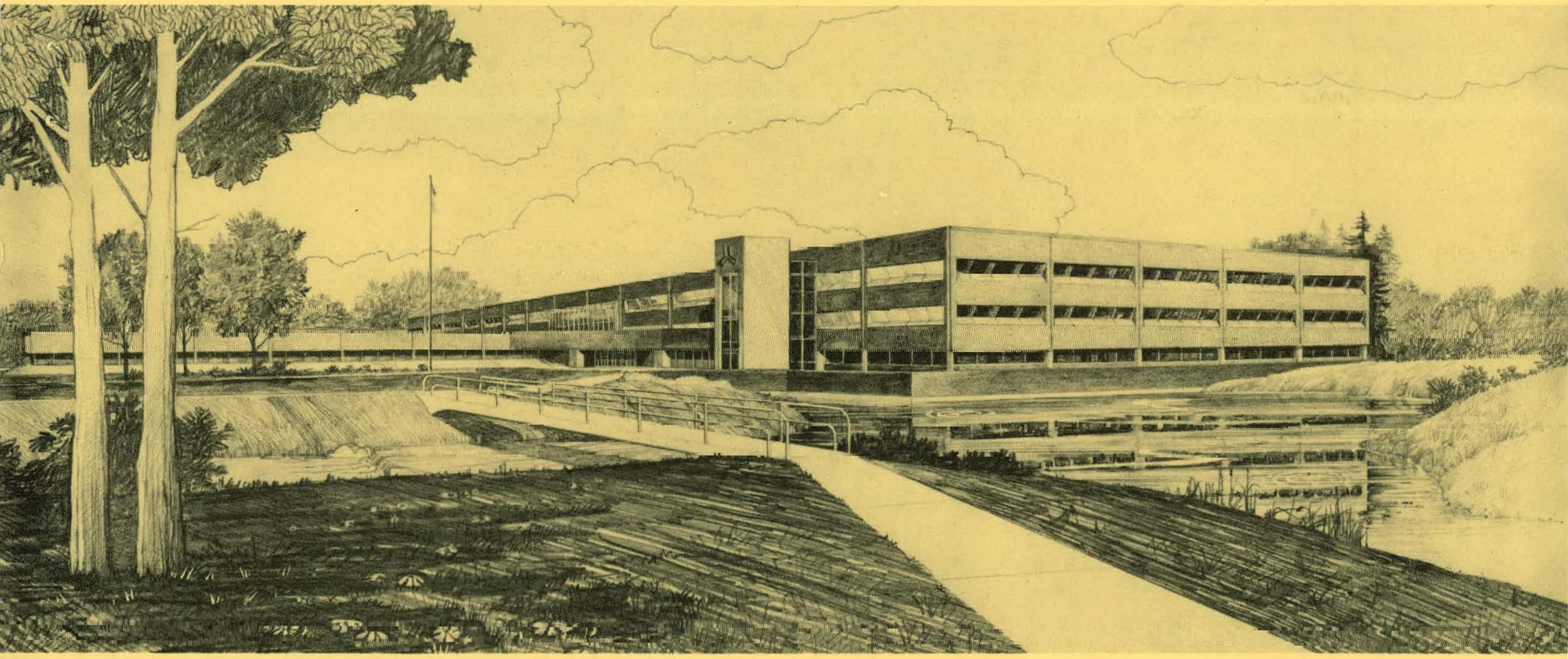

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EG&G Idaho

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SUMMARY

The testing of fusion materials and components in fission reactors will be increasingly important in the future due to the near-term lack of fusion engineering test devices, and the long-term high demand for fusion testing when they do become available. Fission testing is capable of filling many gaps in fusion reactor design information, and should be aggressively pursued. EG&G Idaho has investigated the application of fission testing in three areas, which are discussed in this paper. First, work was performed on the irradiation of magnet insulators. This work is continuing with an improved test environment. Second, a study was performed which indicated that a fission-suppressed hybrid blanket module could be effectively tested in a reactor such as the Engineering Test Reactor (ETR), closely reproducing the predicted performance in a fusion environment. Finally, a conceptual design is presented for a fission-based Integrated Test Facility (ITF), which can accommodate entire first wall/blanket (FW/B) modules for testing in a nuclear environment, simultaneously satisfying many of the FW/B test requirements. This ITF can provide a cyclic neutron/gamma flux, as well as the necessary module support functions.

CONTENTS

FIGURES

SOME APPLICATIONS OF FISSION-BASED TESTING CAPABILITIES IN THE DEVELOPMENT OF FUSION TECHNOLOGY

INTRODUCTION

One primary uncertainty in the design of fusion reactor systems is clearly the effect of the fusion radiation environment on materials and components exposed to it.[1] Materials issues center around radiation-induced changes in such properties as strength, ductility, fracture toughness, and bulk swelling. In addition to these materials questions, issues concerning component suitability arise as engineering development begins. These issues generally involve a complex interaction of several environmental factors. One of these is radiation. The testing required to resolve such questions must therefore involve simulation of multiple effects, including radiation. As engineering development progresses, these tests must examine larger and larger components, culminating eventually in integrated-system experiments involving entire major components, such as blanket modules.

In the past, all material testing for radiation effects has been conducted in the RTNS, other accelerator-based systems, and fission facilities, in the form of coupon-scale irradiations. In the future, some of this testing may be performed in FMIT, and later in INTOR-class fusion machines. However, there will be a continuing, and even increasing need for fission reactor testing, particularly of large test pieces, for three main reasons. First, present design objectives for next-generation devices make them unsuitable as irradiation test facilities. Second, even when suitable fusion facilities are available, the demand for testing will greatly surpass the availability of test space and time, necessitating the reserving of fusion testing for benchmarking the performance of a small percentage of the components of interest. The components tested in fusion reactors must therefore be thoroughly screened and pretested so that the valuable fusion irradiation time is used only for the best components and designs. Such a pretesting program must include the effects of radiation, and therefore must be fission-based. The final reason for the increasing need for fission testing is that it has the capability to fill a gap in fusion reactor design information. This is shown in Figures 1 and 2, which indicate the

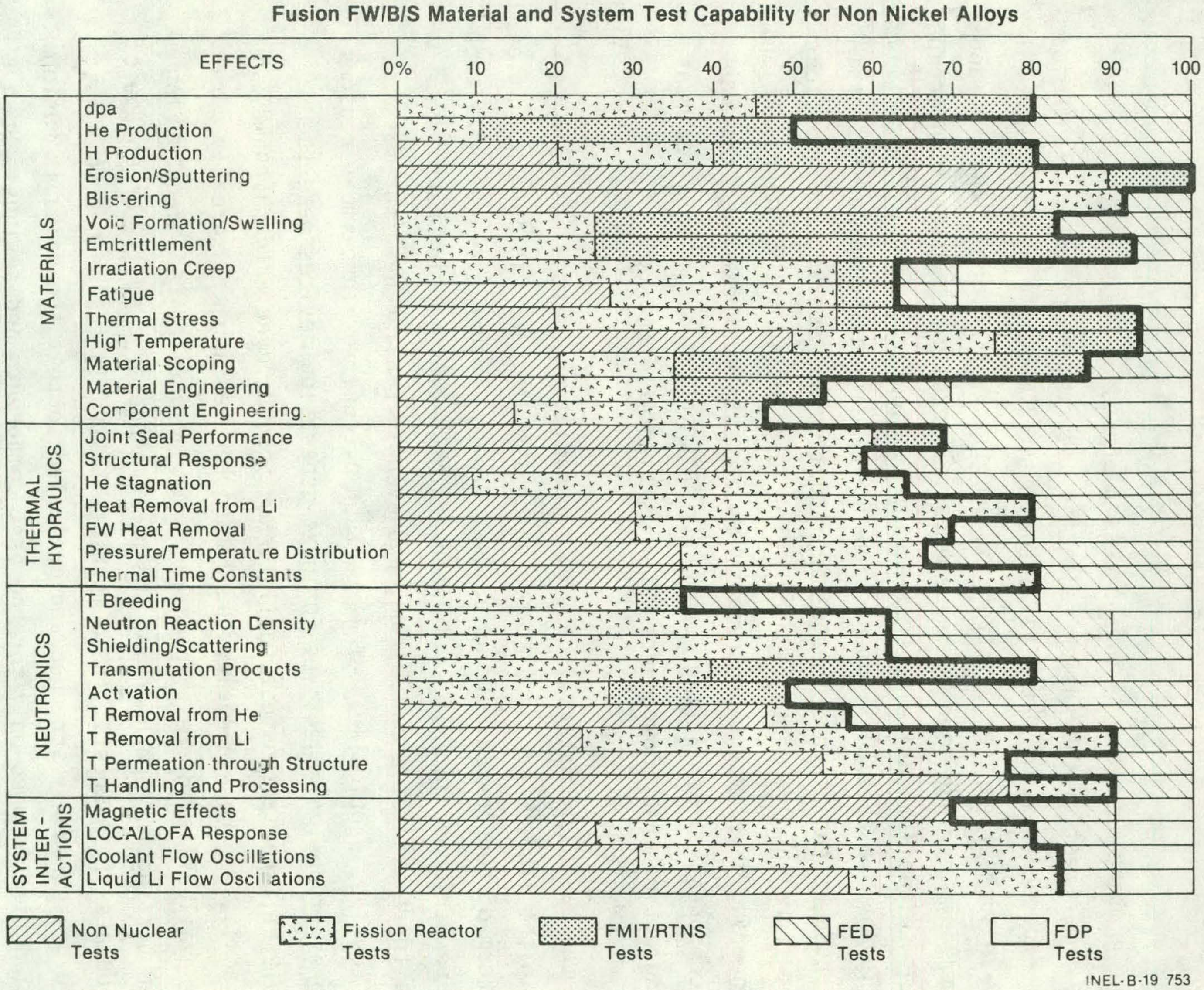

Figure 1. Relative contributions of various technologies to First Wall/Blanket/Shield testing needs for nonnickel-bearing materials.

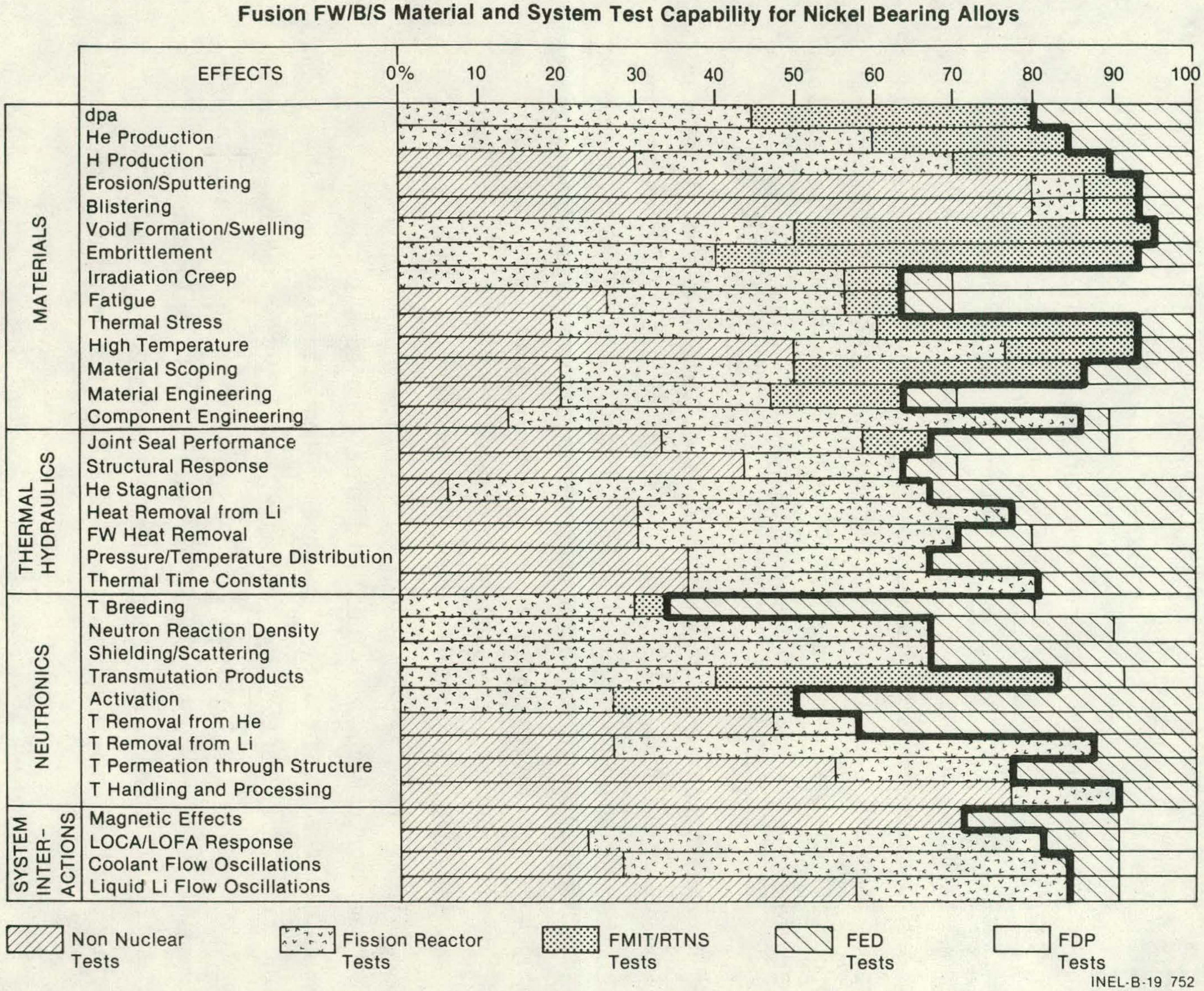

Figure 2. Relative contributions of various technologies to First Wall/Blanket/Shield testing needs for nickel-bearing materials.

degree to which various testing technologies can fulfill the need for information in many important areas. It can be seen that fission testing can provide much of the needed data in many areas. It can also be seen that even if fission testing is aggressively pursued, data shortfalls are possible in several areas. These shortfalls would be seriously worsened by neglecting fission-based testing.

As part of the Nuclear Systems Group of the Technical Management Board of the Office of Fusion Energy, EG&G Idaho was requested to investigate the role of fission test facilities in supporting fusion technology development. A number of factors, including cost effectiveness, and availability, concerning various approaches to nuclear testing of fusion components were addressed. The result of this investigation indicated that the most effective approach from the cost and schedule standpoint is the use of existing fission testing facilities, with modification as required.

EG&G Idaho is active in developing fission testing capabilities for use in the fusion program,[2,3] and in this endeavor has performed studies in three areas which are discussed below. First, work was performed to study radiation damage to magnet insulators. This involved irradiating and performing postirradiation fatigue testing on disks of several organic insulator materials. Second, the nuclear testing of a fission-fusion hybrid blanket was examined. This study addressed the testing, in a light-water reactor (LWR), of the fission-suppressed hybrid blanket concept for the Tandem Mirror Hybrid Reactor. The third area studied was the development of a conceptual design for a fission-based Integrated Test Facility, using the Engineering Test Reactor as an example. A conceptual design was developed for two cases. The first case allows the testing of cylindrical blanket modules similar to the ORNL-TNS concept, with only minor modification of the ETR itself. The second case involves more extensive reactor modification, but allows testing of most prototypical reactor-scale blanket modules.

In the following sections, the three studies mentioned above are discussed. The full details of these investigations can be found elsewhere.[4,5,6]

MAGNET INSULATOR IRRADIATION

The Advanced Test Reactor (ATR) was first used for magnet insulator irradiations two years ago. Location of the irradiation space in the reactor was selected to minimize gamma heating and temperature rise of the insulators. A thermal bond (aluminum powder) is used in the irradiation capsule to maintain specimen temperature to within 1 K of the reactor process water temperature of 325 K. For the irradiation that was performed, the neutron spectrum was well moderated. Eighty eight percent of the resulting damage dose to the material was from gamma radiation and 12% was from the neutrons. In an operating fusion reactor the damage dose is expected to be more nearly balanced between gammas and neutrons.

Insulator materials, which would be applied in a Bitter plate magnet configuration, were irradiated to a dose of 4×10^9 Gy. Insulator G-10 performed well in postirradiation compressive fatigue tests, although a change in the material's resistance from 2×10^{14} ohms to 3×10^5 ohms raises questions about its applicability to this high dose. Additional work on magnet insulator materials is reported in Reference 4.

In order to obtain a more balanced damage dose, insulator irradiations are currently proceeding in the same reactor location that is being used for fuel plate development.[7] The irradiation assembly includes fuel plates arranged in a square array with the insulator capsule positioned at the center of the square as depicted in Figure 3. With this arrangement, the fuel plate assembly serves to convert the well-moderated neutron spectrum to a harder spectrum at the capsule position. The projected damage dose distribution for this setup is expected to be 69% from gamma radiation and 31% from neutrons. Neutron flux monitors in the capsule will determine the actual fluence of both thermal and fast neutrons in each irradiation. The gamma dose is virtually unaffected by the fuel plate assembly while the neutron dose is increased by the change in the neutron energy spectrum. The relative contributions to the damage dose could be modified by such measures as placing a gamma absorber around the capsule. However, that would limit the available space for the capsule.

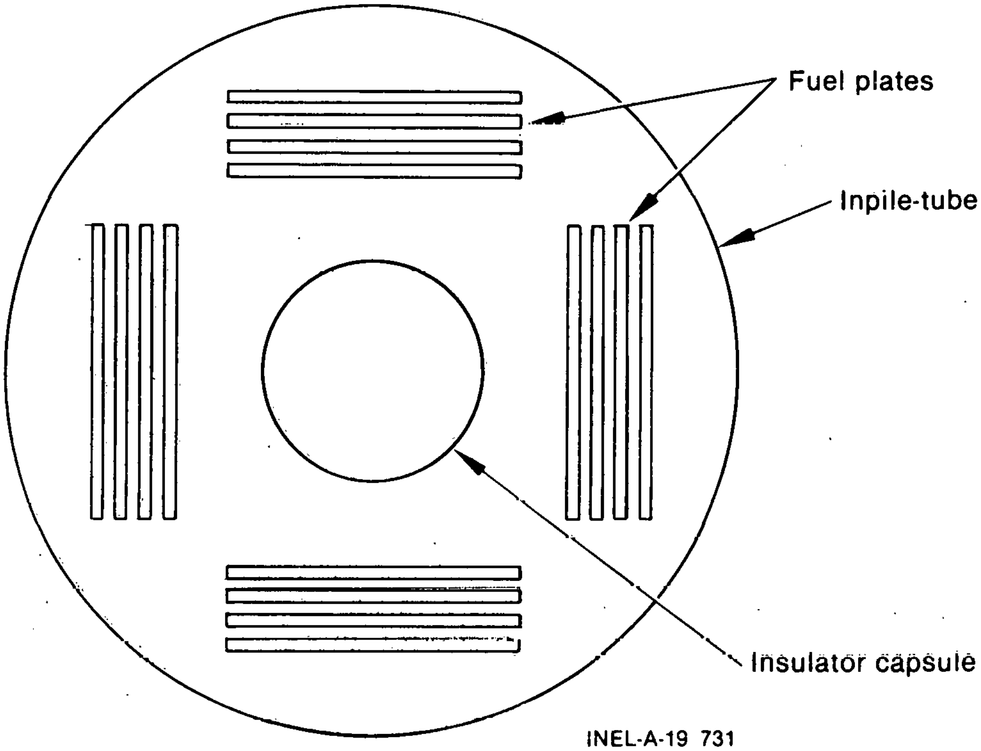

Figure 3. Schematic cross section of fuel plate/insulator irradiation capsule.

FISSION/FUSION HYBRID BLANKET STUDIES

Fission/fusion hybrid reactors are intended to produce fissile fuel for power-producing light-water reactors (LWRs) by capturing fusion-produced neutrons in fertile material in the blanket. To convert fertile isotopes such as ^{232}Th and ^{238}U to a fissile form (through the capture reactions shown in Figure 4) for use in an LWR, the material should be exposed to a flux of thermal and epithermal neutrons. The cross section for neutron capture is higher for low neutron energies. If the fissile isotopes, ^{233}U and ^{239}Pu, produced from the fertile material, are left in the moderated

$$^{232}\mathrm{Th}\,(n,\gamma)\,^{233}\mathrm{Th} \xrightarrow[T_{1/2}\,=\,22.2\ \mathrm{m}]{\beta^-} {}^{233}\mathrm{Pa} \xrightarrow[T_{1/2}\,=\,27.0\mathrm{d}]{\beta^-} {}^{233}\mathrm{U}$$

$$^{238}\mathrm{U}\,(n,\gamma)\,^{239}\mathrm{U} \xrightarrow[T_{1/2}\,=\,23.5\ \mathrm{m}]{\beta^-} {}^{239}\mathrm{Np} \xrightarrow[T_{1/2}\,=\,2.35\mathrm{d}]{\beta^-} {}^{239}\mathrm{Pu}$$

Figure 4. Fissile material breeding reactions.

neutron environment long enough, thermal fission will occur, converting usable fuel material into undesirable fission by-products. The undesirable by-products, most of which are radioactive, produce considerable decay heat, even after the reactor is shut down, and therefore present a serious safety hazard. The concept of the suppressed fission hybrid calls for conversion from the fertile to fissile form in a moderated neutron spectrum, followed by removal of the fissionable material on a time scale which minimizes fission, but is economically viable.

These arguments suggest that the hybrid blanket should be positioned where it can take advantage of the low-energy portion of the neutron spectrum. If this blanket is located immediately behind the first wall in a fission-fusion hybrid reactor, substantial fast-fission would occur, resulting from the high-energy neutron flux. This is illustrated in Figure 5, which shows that the ratio of cross sections (fission to capture) becomes greater than one for neutron energies greater than approximately 1 MeV. The ratio changes quite rapidly with increasing neutron energy, and at 14 MeV the fission rate exceeds the capture rate considerably.

Since the suppressed-fission blanket in a hybrid fusion reactor must be positioned in a low-energy neutron spectrum in order to perform well, these blanket designs should be amenable to testing in an LWR, with its characteristic thermal spectrum. To examine the feasibility of this concept, the two-zone fission/fusion hybrid benchmark blanket design, developed by TRW Inc. for the Tandem Mirror Hybrid Reactor (TMHR), was used.[8] Details of this design are shown in Figure 6. A slab-geometry model of this design (Figure 7) was used to simulate a test of the blanket in the Engineering Test Reactor (ETR) at the Idaho National Engineering Laboratory.

A coupled 121-group (100n, 21γ) library[9] was used to calculate the fluxes, reaction rates, and heating profiles. The results of the calculations for a TMHR first wall/blanket module placed in ETR, were compared with the results for the same module using a fusion neutron source. At an operating power of 175 MW, the ETR is predicted to yield heating rates and tritium production similar to those expected in the TMHR, operating at a neutron wall loading of 1 MW/m^2.

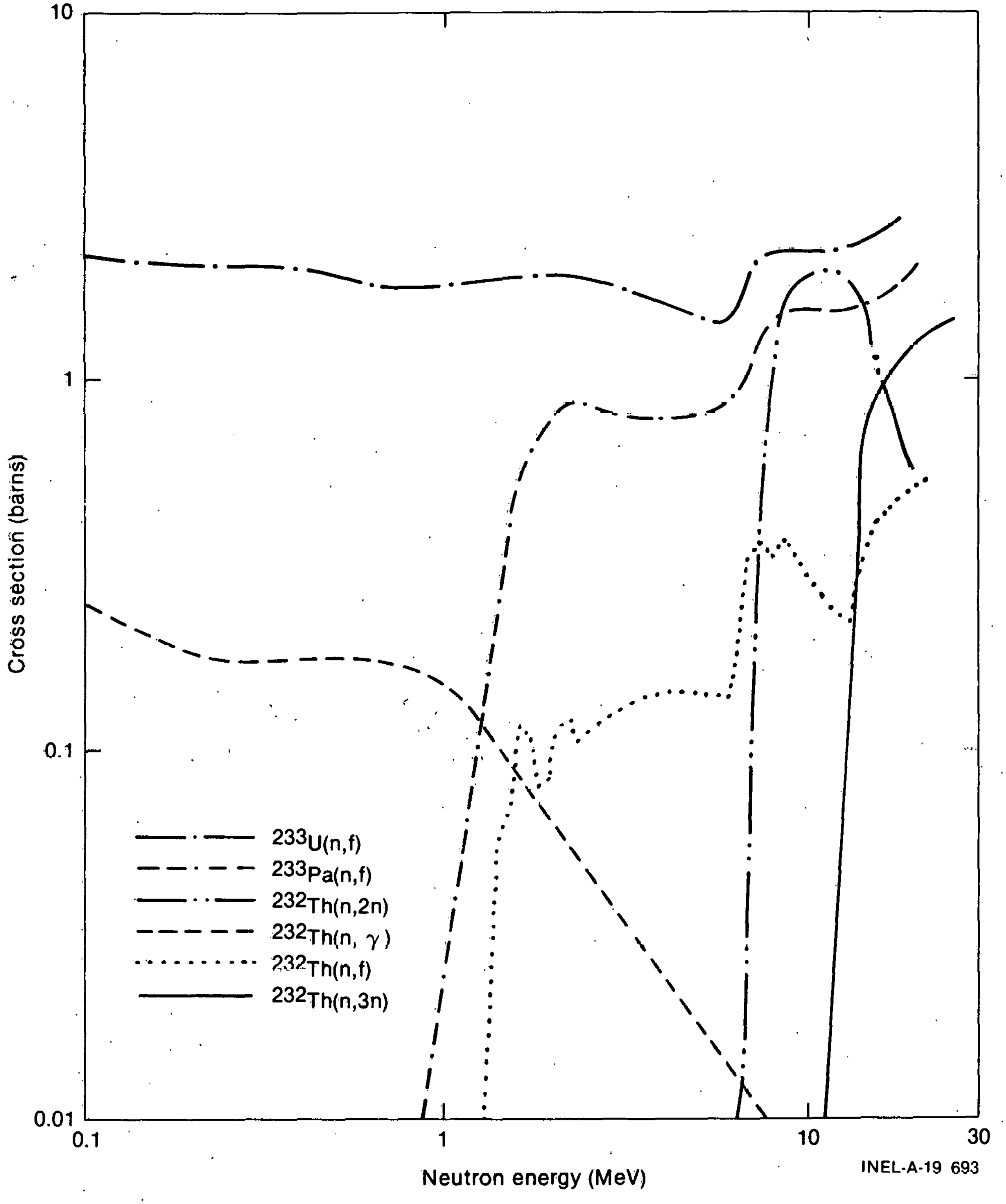

Figure 5. Neutron cross sections pertinent to the ^{233}U fuel cycle.

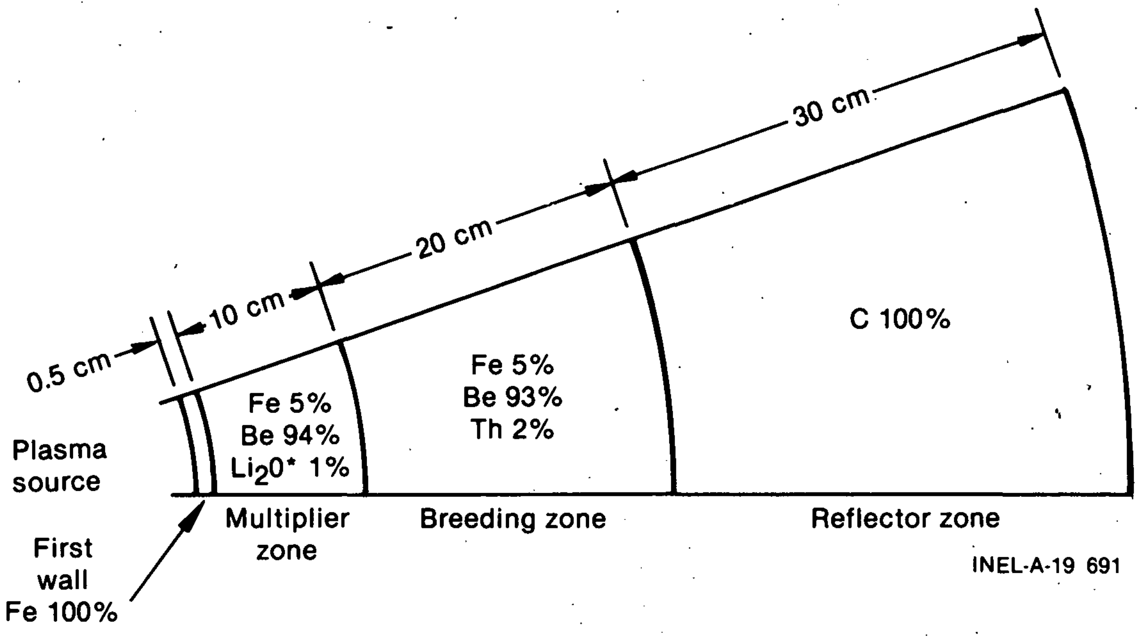

Figure 6. TMHR benchmark hybrid blanket model.

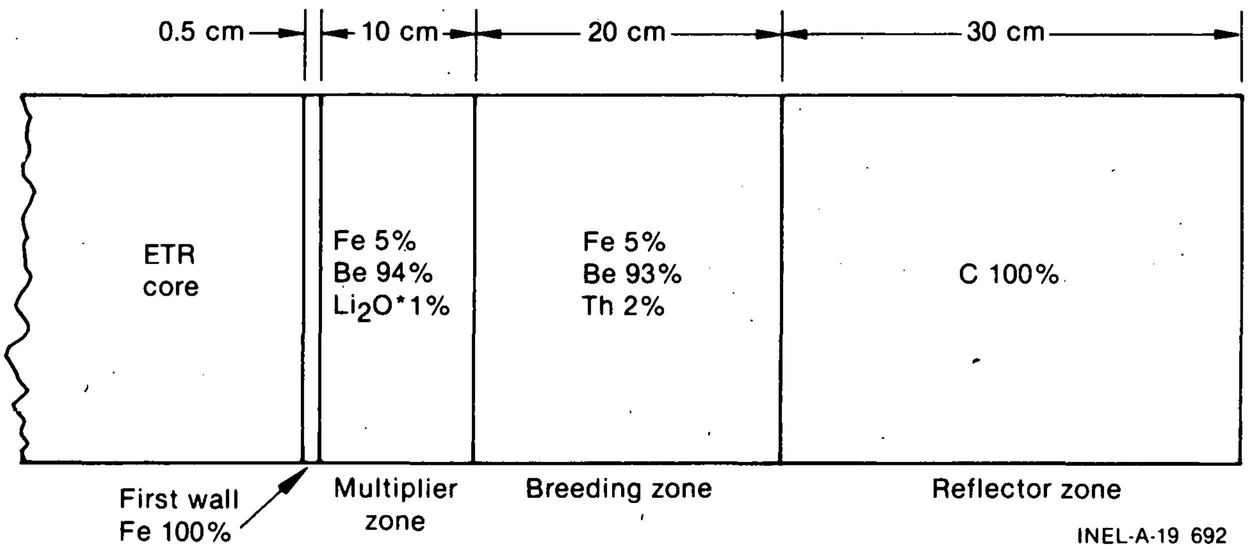

Figure 7. ETR hybrid blanket test model.

The TMHR benchmark design was modelled in infinite cylindrical geometry with a fusion source radius of 170 cm, and the TMHR first wall/blanket module in ETR was modelled in slab geometry with an ETR core thickness of 83.50 cm.

The kerma factors, tritium production factors, and the absorption and fission cross sections extracted from MACKLIB-IV[10] were used with the fluxes computed by the ANISN code[11] to calculate reaction and heating rates. However, the thermal group (group 100) kerma factors for neutron heating and tritium breeding were based on the cross sections given in Reference 9.

The heating rates for TMHR indicate that neutron heating is predominant in the tritium breeder and the first third of the U-233 breeder, while gamma heating dominates in the first wall, in the latter part of the U-233 breeder, and in the reflector. On the other hand, the heating rates for the TMHR module, placed in the ETR core, show that the heating is primarily due to gamma radiation. The heating rate profiles for the ETR and TMHR cases are shown in Figure 8. The heating rate is nearly the same for the ETR and TMHR cases in all zones, except in the first wall. The general trend of the integrated breeding rates in each zone is the same for both the ETR and TMHR cases.

Comparison of the reaction rates of the TMHR benchmark design exposed to a fusion source, with reaction rates in ETR, indicates that the total tritium breeding rate differs by 12.4%. The capture and fission rates in ETR for ^{232}Th are lower by factors of 1.5 and 10, respectively, since the flux is softer in the ETR than in the TMHR case. As expected, the neutron spectra in the tritium breeder (lithium zone) and ^{233}U breeder (thorium zone) are about the same for the two cases except at high energies.

Thus, the ETR and TMHR cases are comparable at an ETR power of 175 MW and a TMHR neutron wall loading of 1 MW/m^2 for the total heating rate profile and the total tritium breeding rate. However, the fission and capture rates are lower in the ETR case.

INTEGRATED TEST FACILITY CONCEPTUAL DESIGN

The objective of this effort was to develop a conceptual design for a fission-based Integrated Test Facility (ITF), in which integrated-system's tests can be performed on various FW/B module designs. These tests will

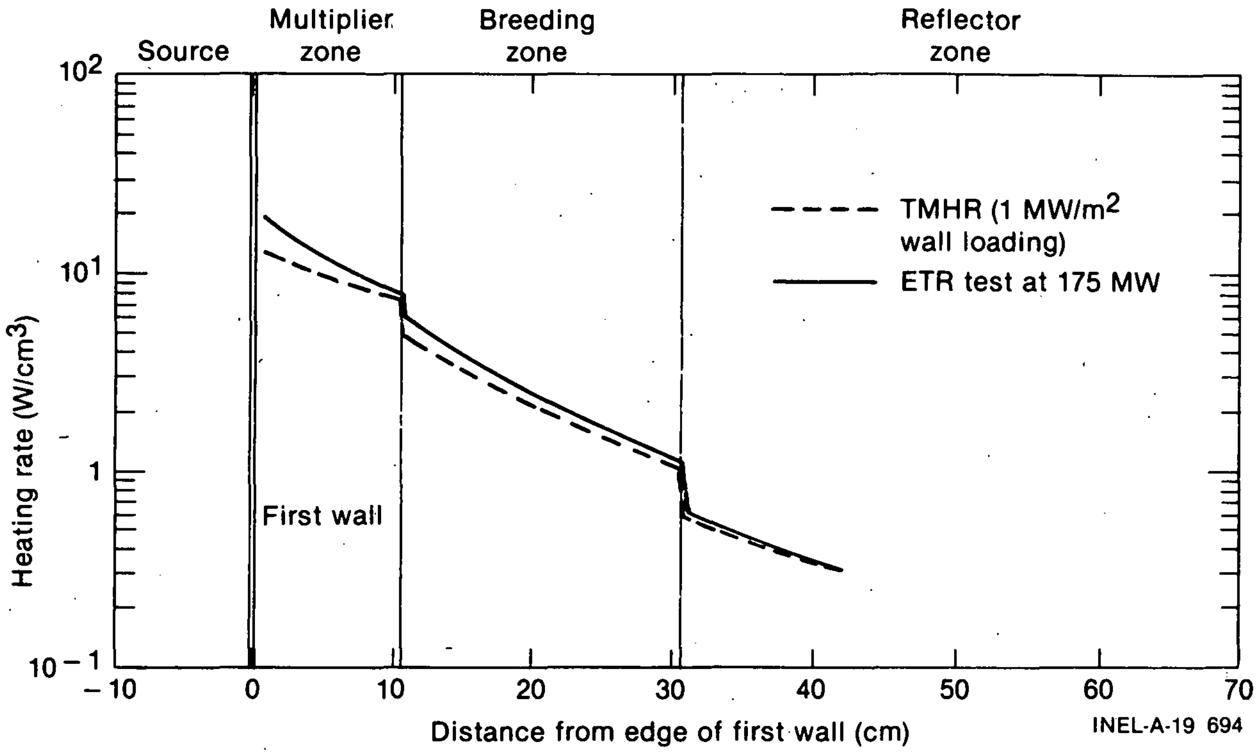

Figure 8. Hybrid benchmark blanket total heating rate comparison.

address many issues, as detailed in Reference 12, including gross thermal performance, lifetime irradiation damage, on-line tritium extraction techniques, fatigue problems, thermal hydraulic/thermomechanical effects, and synergistic effects. In order to address these issues, the ITF must provide many environmental conditions,[12] such as a periodic neutron flux, high vacuum, and magnetic fields. Although considerable difficulty will be encountered in attempting to simulate such an environment, an ITF concept based on a fission reactor has the capability of providing some important factors, provided sufficient volume and power are available.

As an example of how a fission-based ITF might be developed, an existing facility, the ETR, was considered. The large physical size of the ETR, coupled with its high power level (175 MW) makes it attractive for testing slab module designs such as those proposed for INTOR and STARFIRE. In addition, it is well suited for testing cylindrical modules such as the Westinghouse/ORNL module. This module configuration can be tested readily in an existing in-pile tube.

Slab Module Integrated Test Facility

The basic approach taken in the conceptual design of the Slab Module ITF involves vacating a large volume along one side of the ETR core, and installing a test module in the open area. The module would then experience asymmetric bulk heating resulting from the neutron/gamma flux from the core. In order to modulate the neutron/gamma flux to simulate cyclic plasma operation, a shutter is used which captures most of the flux when lowered in front of the module.

The baseline slab module design approach (see Figures 9 and 10) involves modifying the ETR reactor core by removing four rows of fuel elements from the north side of the core, along with the beryllium reflector and associated aluminum core filler pieces. A dry enclosure, which houses the slab module and associated support equipment, piping, and instrumentation, is installed in the vacated space extending from the core grid plate to the top of the reactor vessel flat head. This dry enclosure isolates the module, shutter, and support systems from the reactor core and cooling water, thereby protecting the reactor system in the event of a module failure.

Various support equipment, as required by the module configuration, is located in an ETR basement cubicle and under the biological shielding, adjacent to the reactor vessel. An auxiliary shield/module cask is bolted and sealed to the top of the reactor head. This component is designed to provide radiation shielding for personnel while handling piping connections, and to provide a vehicle for handling the slab module during transport to processing and evaluation facilities.

The module support systems consist of the following: module cooling systems using water or helium, tritium purge systems using helium or circulating liquid lithium, a helium-cooled "shutter" to provide a cyclic neutron flux, and a shutter/module translation system to provide the necessary cyclic movement.

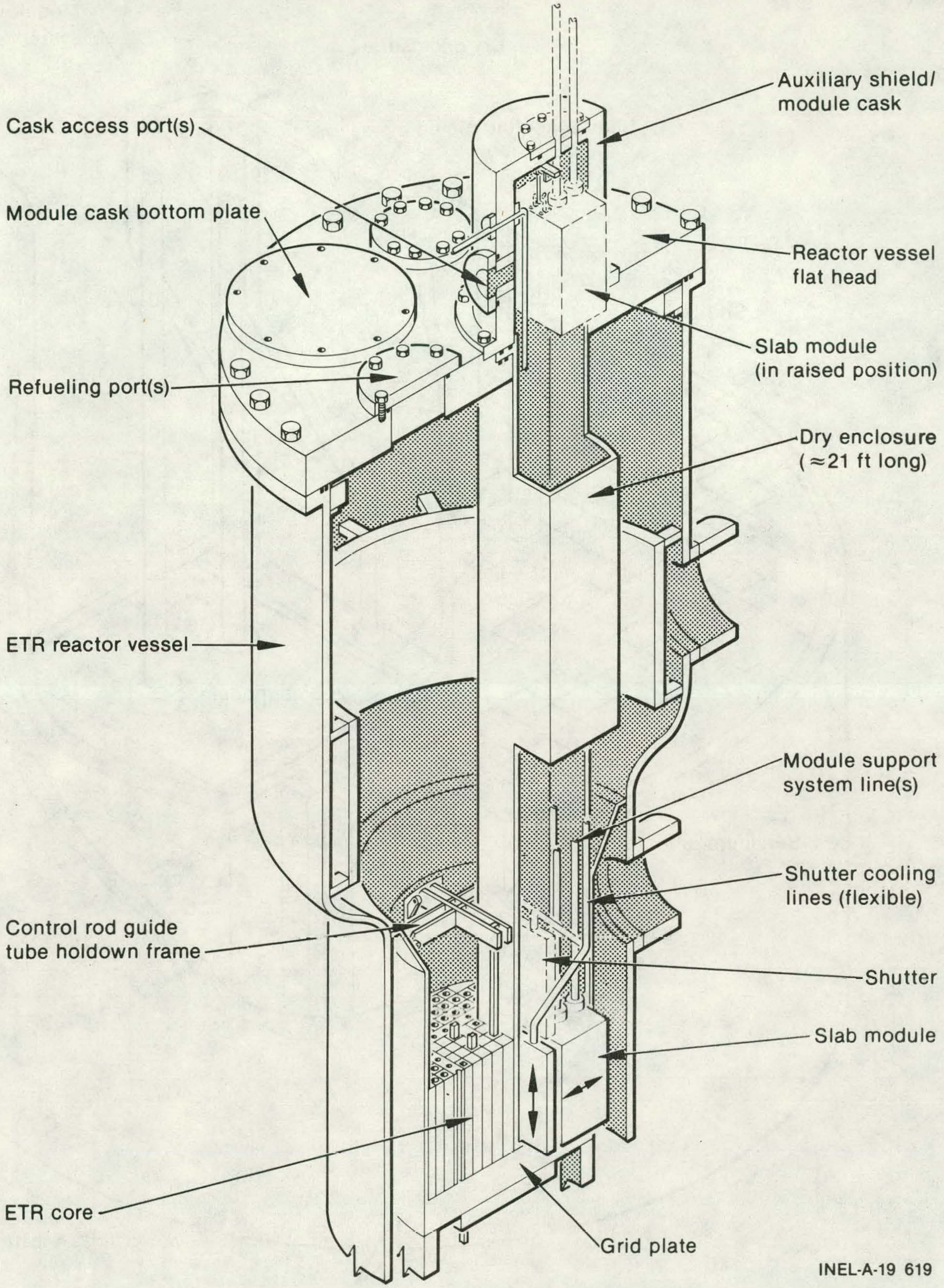

Figure 9. Overall, isometric, cutaway view of Slab Module ITF.

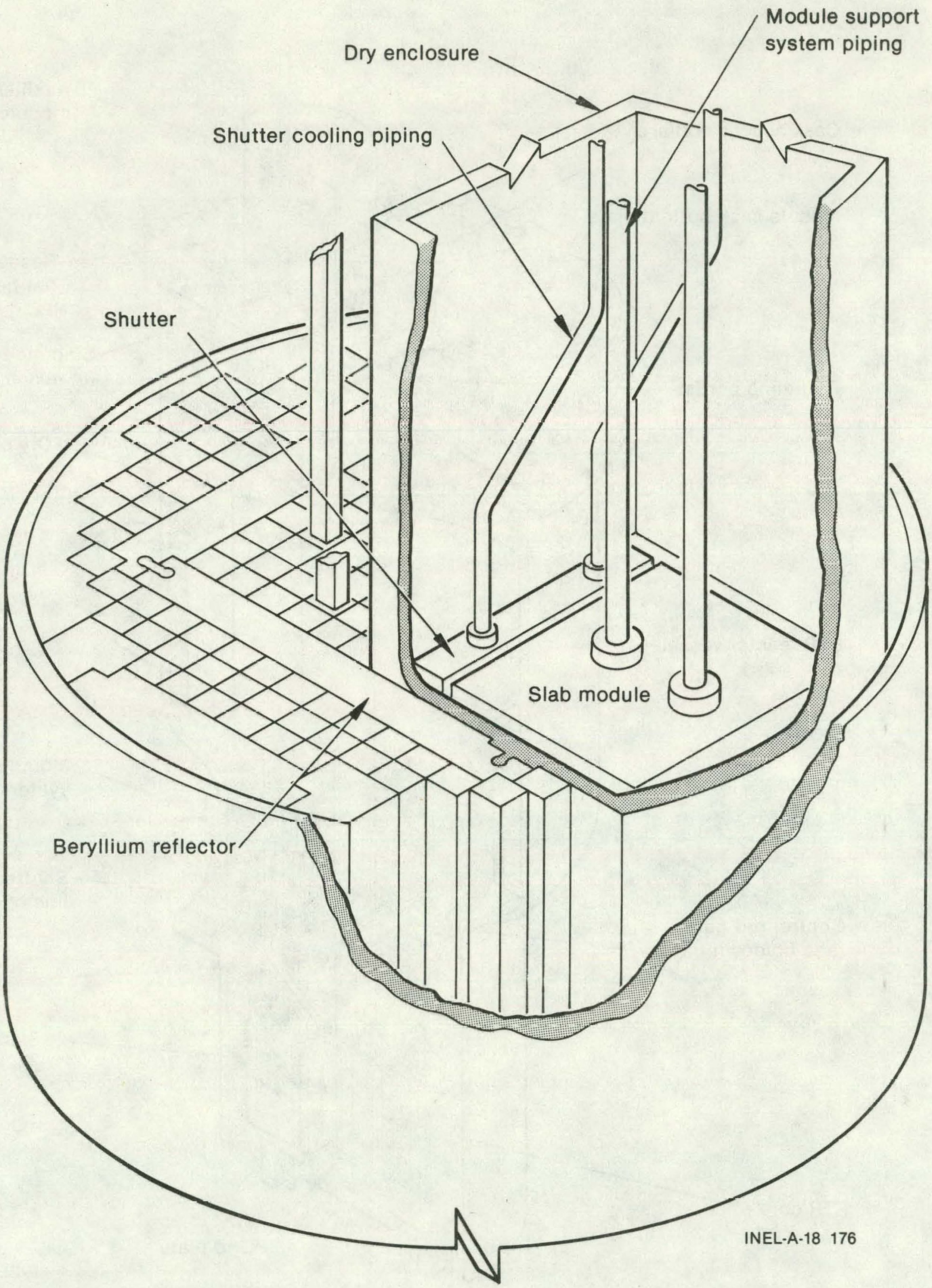

Figure 10. Isometric, cutaway view of Slab Module ITF core area.

The implementation of this ITF concept would allow extensive testing of virtually any type of FW/B module of dimensions up to approximately 1 m on each side. Issues which can be addressed by testing in this type of ITF, include thermal energy recovery efficiency, component lifetime, overall thermal-hydraulic/thermomechanical performance, synergistic effects, and various tritium production and processing concerns. These investigations can be performed over a wide range of environmental conditions, including steady-state cyclic, and off-normal conditions. Such testing will be invaluable in complementing test results from actual fusion devices and in qualifying components for use or testing on them.

Cylindrical Module Integrated Test Facility

The baseline approach for performing cylindrical module tests in the ETR reactor, calls for installing the blanket module in an in-pile tube and inserting this assembly into the core penetration currently used by the Sodium Loop Safety Facility (SLSF) experiment (see Figure 11). This concept requires no internal reactor modifications, with the exception of a lateral support for the in-pile tube. External modifications include the addition of a new reactor vessel flat head (primarily to aid in reactor refueling), and modification of the ETR basement cubicle. The handling and transport of the module will be performed by the existing SLSF Loop Handling Machine (LHM) and Transporter. The major module support systems for the various experiment options are as follows: a liquid lithium circulation system with on-line tritium extraction, a module cooling system using helium, a ^3He first wall heating system, and a thermal cycling system using vertical translation.

Since this ITF concept would involve only minor modification of the reactor itself, the cost would be significantly less than for the Slab Module ITF concept. The same issues could be addressed with either concept, with an equally wide range of possible operating conditions. The only functional difference between the two concepts is that the available test volume in the Cylindrical Module ITF (10 cm in diameter, and approximately 1 m long) is smaller than that in the Slab Module ITF. Either approach, however, would be extremely useful in the testing of FW/B modules.

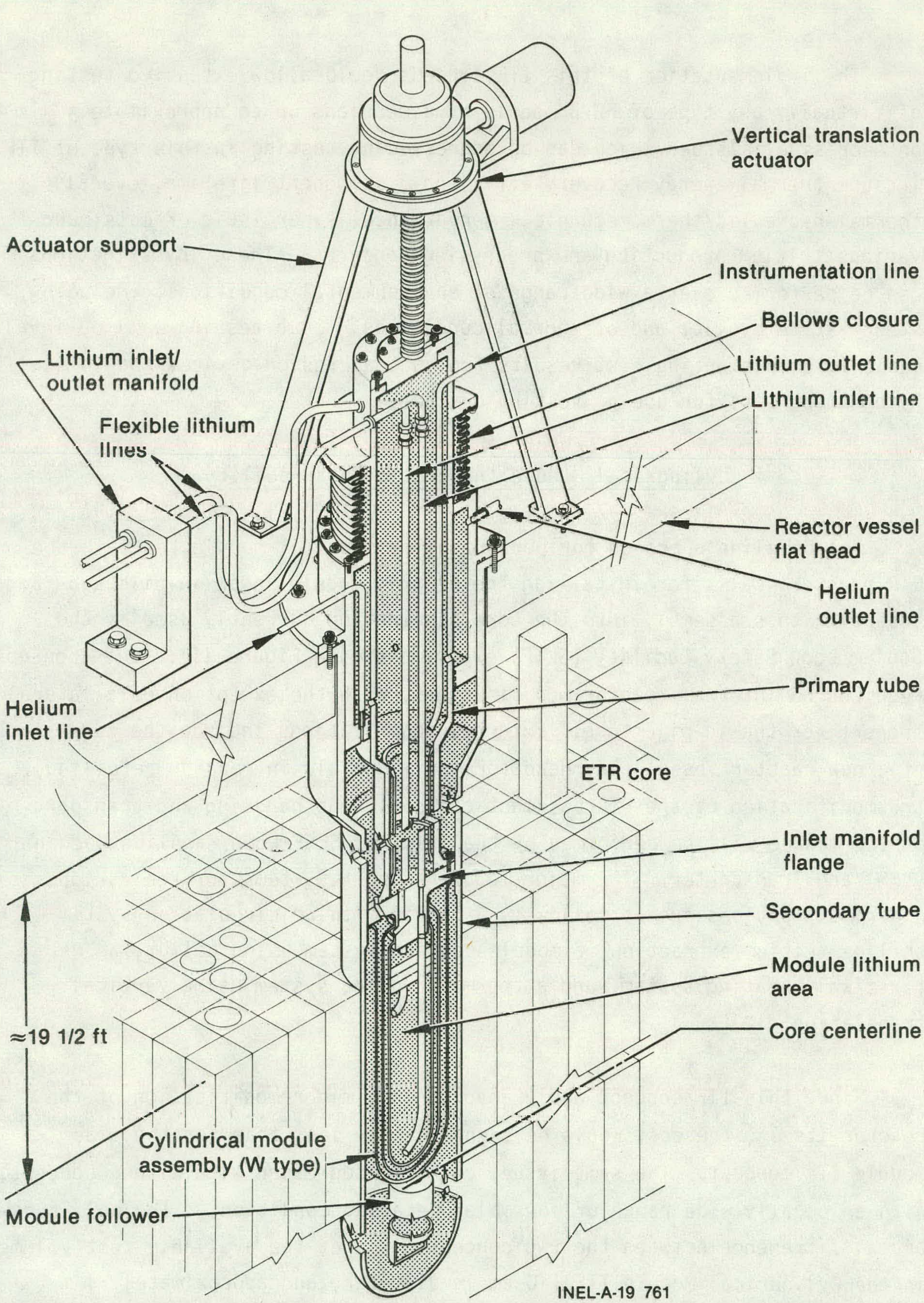

Figure 11. Overall, isometric, cutaway view of Cylindrical Module ITF.

CONCLUSION

Fission testing must play a major role in a cost-effective fusion engineering test program. It is currently in use in materials studies, and will continue to be used for this purpose in the future. However, as engineering development progresses, larger and larger test volumes will be required, eventually leading to fission testing of entire FW/B modules. Fission reactor testing of fission-suppressed hybrid blankets looks particularly attractive in that the expected operation of such a blanket in a fusion reactor is very closely approximated in such a test. Finally, the conceptual designs presented here for an Integrated Test Facility indicate that it is possible to use existing nuclear facilities to conduct full-scale nuclear tests of entire FW/B modules.

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