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ACTIVATION ANALYSIS OF THE COMPACT IGNITION TOKAMAK

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ABSTRACT

The U.S. fusion program has completed the conceptual design of a compact tokamak device that achieves ignition. The high neutron wall loadings associated with this compact deuterium-tritium-burning device indicate that radiation-related issues may be significant considerations in the overall system design. Sufficient shielding will be required for the radiation protection of both reactor components and occupational personnel. A close-in igloo shield has been designed around the periphery of the tokamak structure to permit personnel access into the test cell after shutdown and limit the total activation of the test cell components. This paper describes the conceptual design of the igloo shield system and discusses the major neutronic concerns related to the design of the Compact Ignition Tokamak.

INTRODUCTION

The Compact Ignition Tokamak (CIT) has a major radius of 1.225 m and a nominal deuterium-tritium (D-T) neutron wall loading of 7.8 MW/m^2. An overall description of the conceptual design is provided in a related paper at this conference.[1] The tokamak structure is enclosed in a circular test cell facility with a radius of 10 m. The test cell configuration is shown in Fig. 1. The need to permit limited personnel access after one day following shutdown requires the presence of a bulk shield in the design. Figure 2 is a plan view of the CIT device showing the shielding system.

This device is characterized by several features that complicate the design of the shielding system. The design of the shield must minimize the radiation streaming from the numerous penetrations into the plasma chamber and also be compatible with the interface requirements of the tokamak subsystems for machine accessibility. These include the diagnostics, rf modules, fueling, vacuum duct, facilities, and in-vessel remote maintenance operations. The small size of the test cell makes these objectives even more difficult to achieve, particularly with regard to component disassembly and maintenance. These issues will be resolved during the advanced conceptual design phase planned for FY 1986–87.

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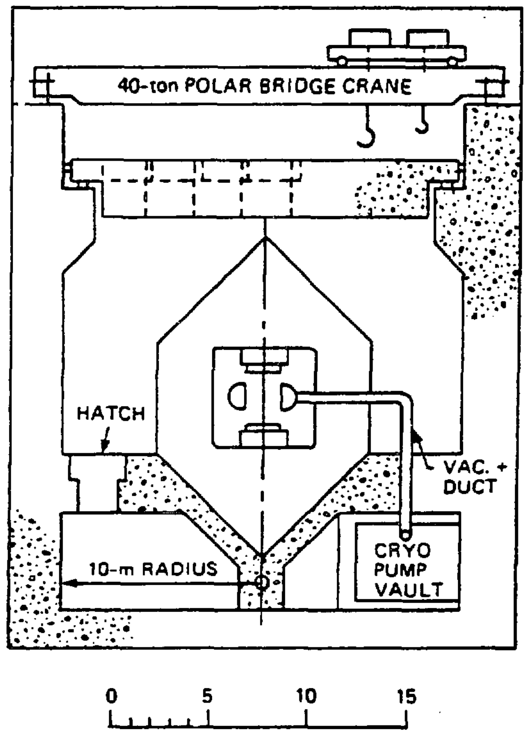

Fig. 1. Test cell configuration of CIT.

SHIELD DESIGN CRITERIA

The basic shield design criteria are dictated by material, biological dose, and dose rate limits. Two functions must be simultaneously served. The first is to protect reactor

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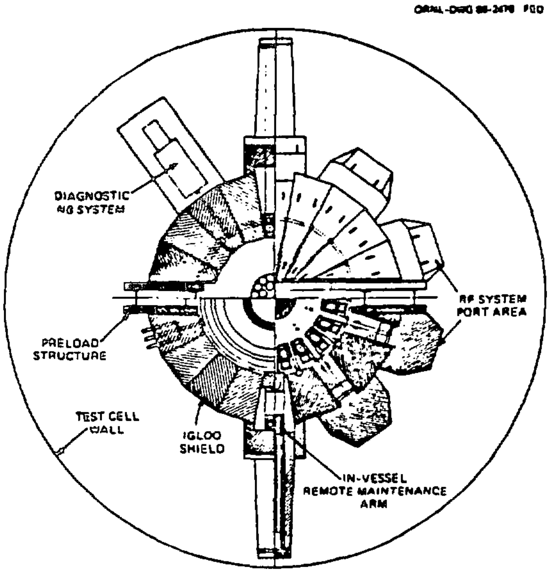

Fig. 2. *Multilevel plan view of CIT shielding system.*

components and subsystems from life-limiting radiation damage levels. In addition to providing local radiation protection, the shielding also serves to reduce the total activation within the test cell, thereby minimizing radioactivity levels at the time of decommissioning.

The second function of the bulk shielding system is to allow personnel access into the test cell facility shortly after shutdown. The criterion adopted for personnel access is to limit the biological dose rate external to the bulk shield to 1 mrem/h at 24 h after shutdown. This is consistent with guidelines for the protection of radiation workers.[2] Radial flux levels and line-of-sight techniques were used to establish the nominal bulk shield thickness. Based on this design criterion, a nominal radial bulk shield thickness of 1.8 m is required around the torus. There will also be local requirements for duct and component shielding and shielding around auxiliary equipment. These, however, will be quantified during the more detailed design phase of the project.

DESIGN DESCRIPTION OF CIT SHIELDING SYSTEM

The shielding system is designed to provide radiation protection for reactor components and personnel within the test cell. The test cell wall, 2.8-m-thick reinforced concrete, limits the biological dose outside the test cell to 1 rem/year during operation. The bulk shield surrounding the tokamak structure facilitates hands-on access after shutdown and minimizes the activation of the test cell walls and components within the test cell located outside the shield. The equipment that may benefit from hands-on access includes the diagnostic equipment, coolant and electric lines, remote handling equipment, and primary test cell crane and work platform. The shield consists of portable blocks of ordinary concrete placed around the periphery of the torus. The close-fitting shield blocks, located approximately 10 cm from the reactor components, are assembled to follow the irregular contours of the 10 port extensions. The blocks have irregular shapes which include monolithic, trapezoidal-shaped, and pie-shaped components with supporting structural pieces. They are designed to be easily stacked and disassembled in order to provide maximum access to the plasma chamber. Each block is designed with several offsets to reduce radiation peaks from neutron and gamma streaming. The maximum size of a block is limited to the 40-ton lifting capability of the primary test cell crane. The detailed design of the shielding will be governed by specific interface requirements of reactor subsystems.

METHODOLOGY OF ANALYSIS

The shutdown dose rate analysis for the bulk shielding system employed one-dimensional transport and activation techniques. This methodology is typical for a conceptual design. The transport calculations were performed using the discrete-ordinates code ANISN with an S8 symmetric angular quadrature set and a P3 legendre expansion for the scattering cross sections. The associated transport cross-section library was derived from ENDF/B-IV and includes 25 neutron groups and 12 gamma groups.[3] The 25-group neutron activation cross-section library was derived from MONTAGE-400,[4] and the decay gamma energies and yields were obtained from the Table of Isotopes. Nuclear responses were computed with the MACKLIB-IV library.[5] The ANISN code[6] was first used to determine the steady-state neutron fluxes along the radial midplane of the torus. These fluxes were then used by the REBATE code[7] to determine the radioactivity produced and the energy and spatial dependence of the decay gamma source generated as a function of time after shutdown. Subsequent gamma transport calculations were performed with the ANISN code to determine the dose equivalent for specific shutdown time intervals.

Previous studies have shown that the neutron wall loading varies in the poloidal direction and is maximum at the outboard midplane. The wall loading is typically 20–30% higher than the nominal value in this region. The CIT has a nominal neutron loading of 7.8 MW/m^2 at the plasma edge based upon a plasma operation producing 300 MW of fusion power and a plasma surface area of 30.7 m^2. The wall loading at the outboard midplane plasma edge is then 9.3 MW/m^2, while the inboard midplane loading is ~95% of the nominal value, or 7.4 MW/m^2. The corresponding neutron source used in the transport calculations was consistent with these wall loadings.

Two one-dimensional geometric configurations were investigated to represent the test cell facility. One model employs toroidal cylindrical geometry and is used to evaluate the nuclear responses on the inboard and outboard midplane sections near the plasma. At larger distances from the plasma, spherical geometry is more appropriate because the flux is treated as being attenuated as $1/r^2$ rather than as

$1/r$ in the cylindrical model. On this basis, spherical geometry was used to determine the dose equivalent outside the bulk shield. As part of the supporting studies for the conceptual design, results from both models were compared and found to be similar at this outboard location.

Table 1 shows the material inventory within the CIT test cell. One source of uncertainty in the activation analysis is associated with the material inventory. The material compositions used in a design analysis may not correspond to the actual composition of the material fabricated and assembled by the manufacturer. This problem, however, is inherent to all predictions of nuclear responses.

The composition of the igloo shield was chosen from economic considerations to be ordinary concrete. Although the use of borated or barytes concrete would incur a significant cost penalty, requirements for machine accessibility may necessitate their use in localized areas around the device. The impact of using these alternate materials has been examined in the studies leading to the conceptual design.

The design value for the nominal bulk shield thickness, 1.8 m, was based upon a midplane radial analysis through the intercoil structure. This structure was modelled as an empty region bounded by two stainless steel beams. Table 1 shows the post-shutdown gamma dose rates at various outboard locations along this midplane radial traverse. A midplane radial analysis through the toroidal field coil plane results in a nominal thickness of 1.4 m.

VARIATION IN OPERATING SCENARIO AND SHUTDOWN TIME

The operating history of the plasma for the scheduled 3000 full-power-shot lifetime corresponds to a 3.7-s pulse width, a 1-h interval between pulses, and operation with eight pulses per day for a total of 375 d. This is followed by a 24-h shutdown cooling period. The models used in the current analysis are restricted to examining regular pulsing sequences. The actual operating history of the plasma will not follow this regular scenario and will be spread over a 5-year period with scheduled as well as unscheduled downtime for maintenance. Variations in the operating time show that after one pulse the dose equivalent at 24 h is 10% of its lifetime value, after operation for 1 d the dose is 66% of its lifetime value, and after operation for one week (5 d) the dose has saturated at its lifetime value. This is due to the buildup of the longer-lived isotopes.

Table 2 lists the important isotopes contributing to the 24-h dose rate, along with the corresponding decay modes and decay gamma energies. If the cooling period could be extended to 48 h, the ^{56}Mn contribution could be eliminated and would result in a nominal shield thickness of 1.7 m. A shutdown time of one week would significantly reduce the ^{64}Cu and ^{24}Na components and would reduce the shield thickness to approximately 1.2 m. Figure 3 shows the variation of the shield thickness required to satisfy the 1-mrem/h

Table 1. Material inventory in the CIT test cell (one-dimensional radial traverse through intercoil structure)

Component	Material	24-h outboard dose rate (mrem/h)
OH solenoid	Cu-102/Inconel-718/insulation	
Inboard TF coil	Cu-102/Inconel-718/insulation	
Thermal insulation	Polyimide	
Vacuum vessel	Inconel-625	
First wall	Graphite	
Plasma		
Scrape-off		
First wall	Graphite	
		2×10^7
Vacuum vessel	Inconel-625	
Insulation	Polyimide	
Intercoil structure	SS-304	
		2×10^5
Foam insulation	Polyimide foam	
Igloo shield	Ordinary concrete	1.0
Test cell wall	Reinforced concrete	

Table 2. Major isotopes contributing to the 24-h dose rate

Isotope	Half-life (h)	Parent	Decay mode[a]	Decay photons Energy (MeV)	Decay photons Yield (%)
^{56}Mn	2.6	^{55}Mn	β^-	2.11	15
		^{56}Fe		1.81	28
				0.85	99
^{64}Cu	12.7	^{63}Cu	β^-, EC, β^+	1.35	0.6
				0.51	39
^{24}Na	15.0	^{23}Na	β^-	2.75	99.9
		^{27}Al		1.37	100
^{57}Ni	36.1	^{58}Ni	EC, β^+	1.92	14.7
				1.76	7.1
				1.38	77.6

[a]Decay modes: β^-, beta emission; β^+, positron emission; and EC, electron capture.

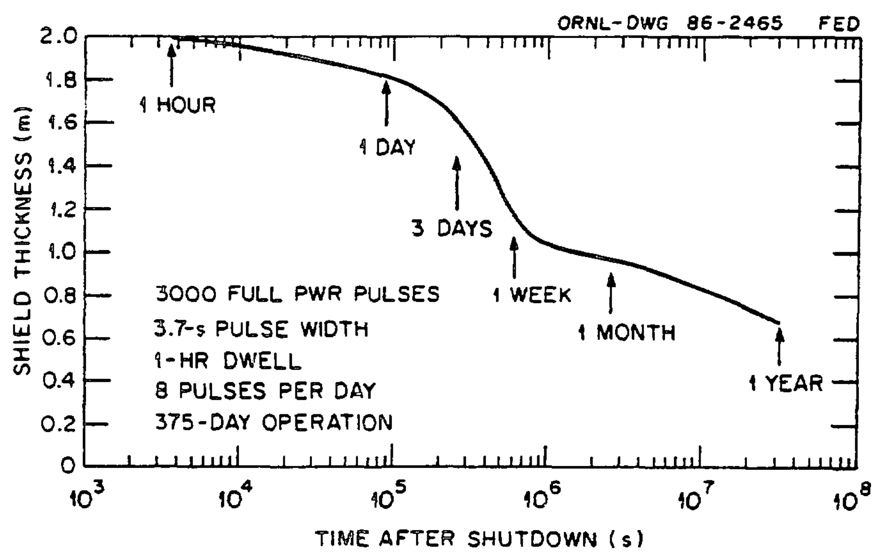

Fig. 3. Variation of shield thickness with time after shutdown.

criterion with time after shutdown. Much of the ^{24}Na in the test cell is emitted from the concrete shield. The utilization of low-sodium concrete could be considered but may result in a significant cost penalty.

MAJOR NEUTRONICS CONCERNS IN THE CIT DEVICE

The major neutronics concerns associated with the operation of the CIT device are highlighted in Fig. 4. The most critical are the numerous penetrations, which include the vacuum duct, rf, fueling, diagnostic, and remote maintenance ports. Gaps and inhomogeneities also play an important role in increasing the radiation response levels due to neutron streaming. These radiation peaks can result in nonuniform nuclear heating, atomic displacements, and gas production rates. Shielding must be provided to isolate the test cell and diagnostic equipment from the effects of these penetrations. The shielding requirements must then be iterated with interface requirements of tokamak subsystems such as those for machine accessibility. It may be desirable to incorporate the use of high-density concrete, such as borated or barytes concrete, in localized areas where access requirements become restricted. The penetration shield design will be developed during the advanced conceptual design phase of the project.

The additional concerns identified in Fig. 4 include radiation damage to the inboard thermal insulation and to the diagnostic equipment. The neutron flux near the vacuum vessel is $\sim 10^{19}$ neutrons/(m^2-s), and the total dose for the 3000-pulse lifetime is $\sim 10^8$ Gy. This requires the use of polyimide rather than epoxy G-10 for the thermal insulation. In addition, some of the diagnostic equipment will require local shielding to minimize radiation-induced noise.

Other radiation concerns that require near-term quantification include determination of the lifetime of the hydraulic assembly with respect to radiation damage and a reevaluation of the thickness required for the test cell walls and roof. The thickness of 2.8 m for the walls and roof was established with no credit given for the CIT igloo shield. The effects of skyshine on the requirements for the roof thickness and the activation of the test cell components outboard of the bulk shield will also be considered. Analyses of the activation of the test cell air atmosphere and of the corrosion products and ^{14}C contained within the coolant streams are needed as input for a reactor safety assessment. The ^{14}C is produced within the liquid nitrogen coolant as a result of an (n,p) reaction with ^{14}N. Finally, since the device is planned to operate with D-D prior to the ignited D-T operations, an assessment of the shield requirements for D-D operation will be performed to investigate the feasibility of hands-on access. Subsequent to these near-term design tasks will be a determination of the component radioactivity levels at the time of decommissioning. This will quantify the waste disposal characteristics of the device.

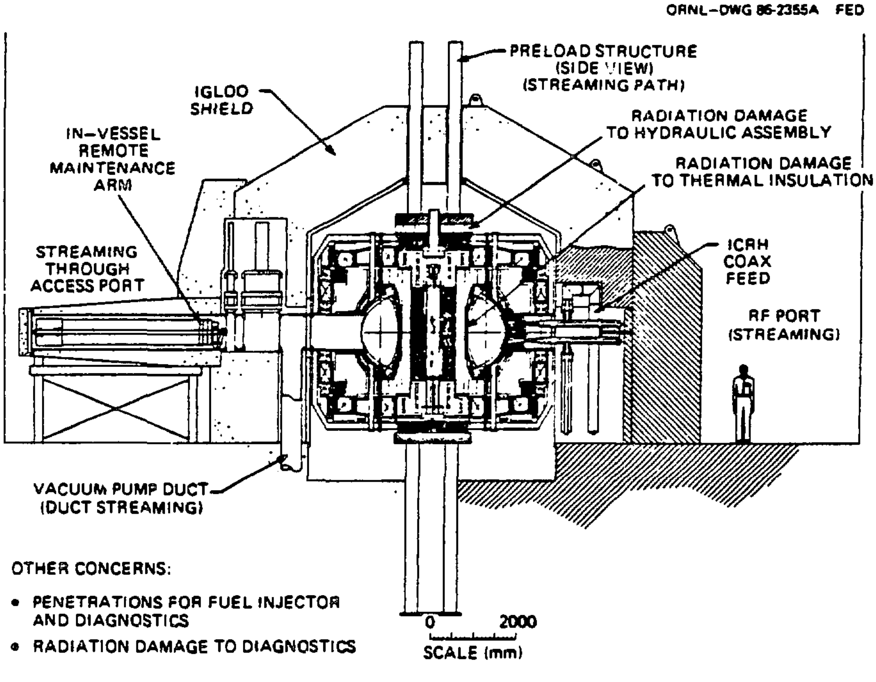

Fig. 4. Side elevation view showing major neutronics concerns.

SUMMARY AND CONCLUSIONS

In summary, a conceptual design for the igloo shielding system surrounding the CIT has been completed. A nominal radial shield thickness of 1.8 m has been established based upon a criterion to limit the biological dose rate external to the bulk shield to 1 mrem/h at 24 h after shutdown. The most critical radiation concern in the CIT relates to the numerous penetrations in the device. The shield must be designed to minimize radiation streaming and also be compatible with tokamak access requirements for the reactor subsystems. It may be desirable to incorporate higher density concrete in localized areas of the igloo shield concept. To accomplish these objectives within the small test cell facility is a challenging design problem. During the next phase of the project, the shield design will be developed in sufficient detail to quantify the local shield requirements around the torus and auxiliary equipment. Optimization of the test cell walls and roof, radiation damage to specific components, activation of the test cell air and coolant streams, and waste disposal characteristics will also be determined.

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REFERENCES

1. C. A. FLANAGAN et al., "Overview of the Compact Ignition Tokamak," presented at the Seventh Topical Meeting on the Technology of Fusion Energy, Reno, Nevada, June 16–19, 1986.
2. "Environmental Protection, Safety, and Health Protection Program for DOE Operations," DOE order 5480.1, U.S. Department of Energy (1981).
3. L. P. KU and J. G. KOLIBAL, "PRAX-C, PPPL Radiation Analysis Cross Sections for the Cray," EAD-R-1, Princeton Plasma Physics Laboratory (1982).
4. "Montage-400, 100-Group Neutron Activation Cross Section Data for Fusion Reactor Structure and Coolant Materials," DLC-33, Radiation Shielding Information Center, Oak Ridge National Laboratory (1979).
5. "MACKLIB-IV, A Library of Nuclear Responses Generated with the MACK-IV Computer Program from ENDF/B-IV," ANL/FPP/TM-106, Argonne National Laboratory (1978).
6. L. P. KU and J. G. KOLIBAL, "ANISN/PPL-C, A One-Dimensional Multigroup Discrete Ordinates Code, A User's Guide for the Cray Version," EAD-R-11, Princeton Plasma Physics Laboratory (1982).
7. L. P. KU and J. G. KOLIBAL, "REBATE/PPL-C, A One and Two Dimensional Activation Analysis Code," EAD-R-4, Princeton Plasma Physics Laboratory (1982).

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PRESENTED TO:

SEVENTH TOPICAL MEETING
ON THE
TECHNOLOGY OF FUSION ENERGY

AMERICAN NUCLEAR SOCIETY

RENO, NEVADA
JUNE 19, 1986

OVERVIEW

A conceptual design of the Compact Ignition Tokamak (CIT) has been completed

Major nuclear characteristics:

- **Major radius: 1.225 m**
- **Nominal D-T neutron wall loading: 7.8 MW/m^2**
- **Ignited operation:**
  3.7-sec pulse width
  3000 full power pulses

An igloo shield surrounds the periphery of the tokamak structure

SCOPE

A nuclear analysis of the CIT is in progress

- **Conceptual design of the igloo shield system**
- **Detailed shield design and activation analysis**

Test cell configuration of CIT

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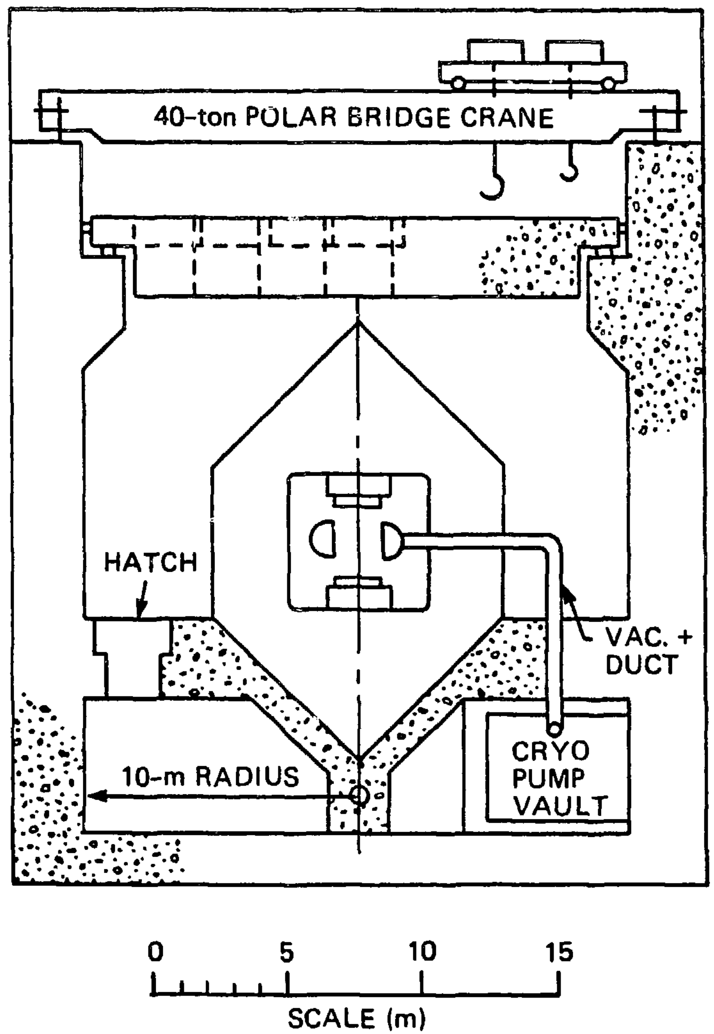
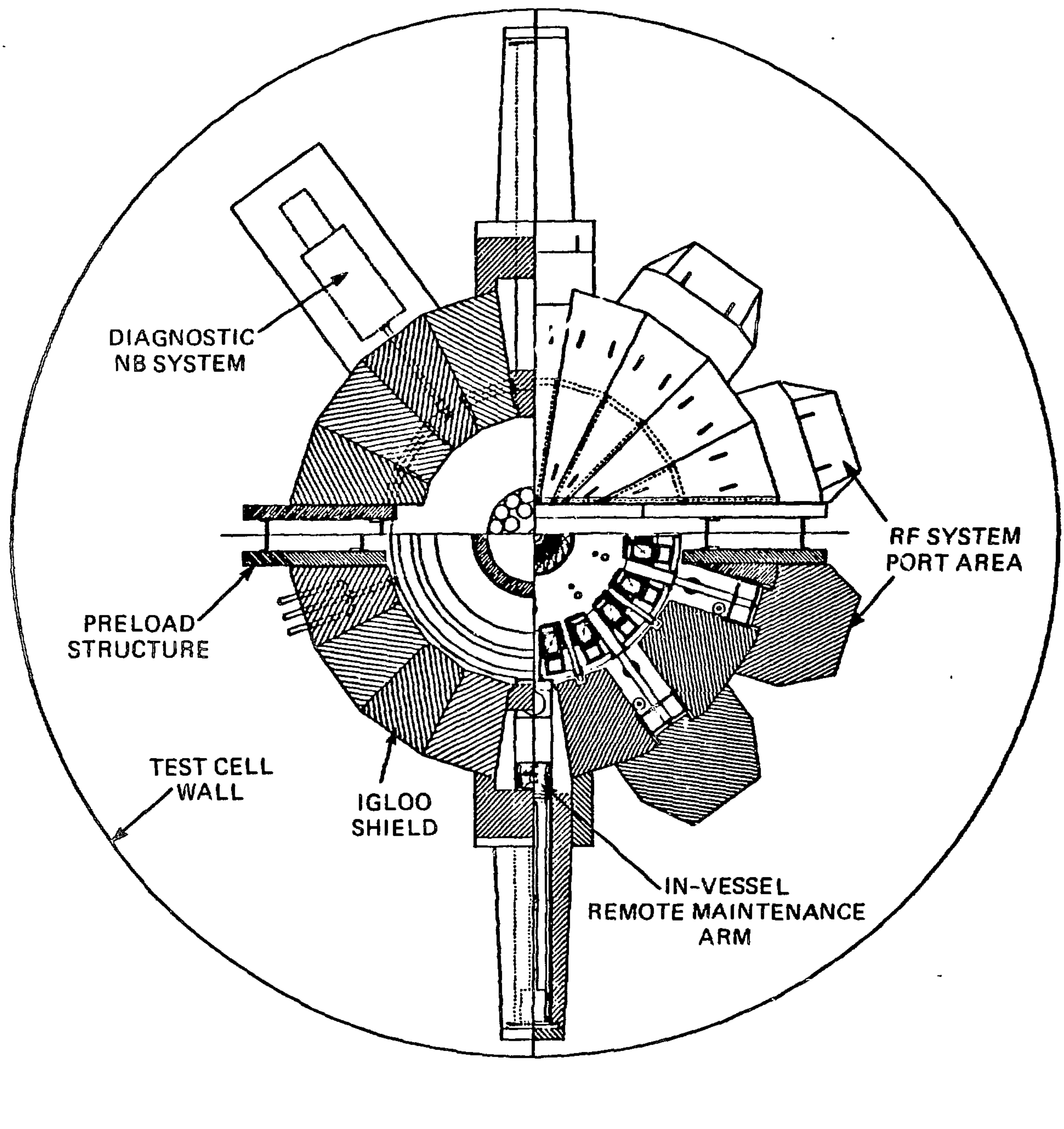


Multilevel plan view of CIT shielding system



DESIGN DESCRIPTION OF IGLOO SHIELD

The shielding system is designed to provide radiation protection for reactor components and personnel within the test cell

- Nominal thickness: 1.8 m
  based on 1 mrem/h criterion at 24 h after shutdown outside of bulk shield
- Composition: ordinary concrete
- Configuration:
  portable blocks placed around periphery of tokamak structure
  - size of blocks limited to 40-ton capability of test cell crane
  - shape of blocks permit close-fit and ease in assembly and disassembly - maximizes access to plasma chamber
  - design of blocks minimizes streaming

POST-SHUTDOWN DOSE RATE ANALYSIS

One dimensional transport and activation techniques were used for the conceptual design of the igloo shield

- **1-D geometric representation of test cell:**
  - **spherical**
  - **toroidal cylindrical**
- **neutron transport using ANISN**
  - **ENDF/B-IV**
- **activation using REBATE**
  - **Montage-400**
- **gamma transport using ANISN**
- **dose equivalent rate**

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MATERIAL INVENTORY IN THE CIT TEST CELL

Component	Material	Major Isotopes
OH sol & Inboard TF	Cu-102/ Inc-718/Ins	Cr-51, Mn-54, Ni-57, Co-58, Co-60, Cu-64, Nb-92, Mo-99
Outboard TF	Cu-102/Ins	Co-60, Cu-64
Coil casing & intercoil structure	SS-304	Cr-51, Mn-54, Mn-56, Ni-57, Co-58, Fe-59, Co-60
Vacuum vessel	Inc-625	Cr-51, Mn-54, Mn-56, Ni-57, Co-58, Co-60, Nb-92, Mo-93m, Mo-99
Igloo shield	Ordinary concrete	Na-24

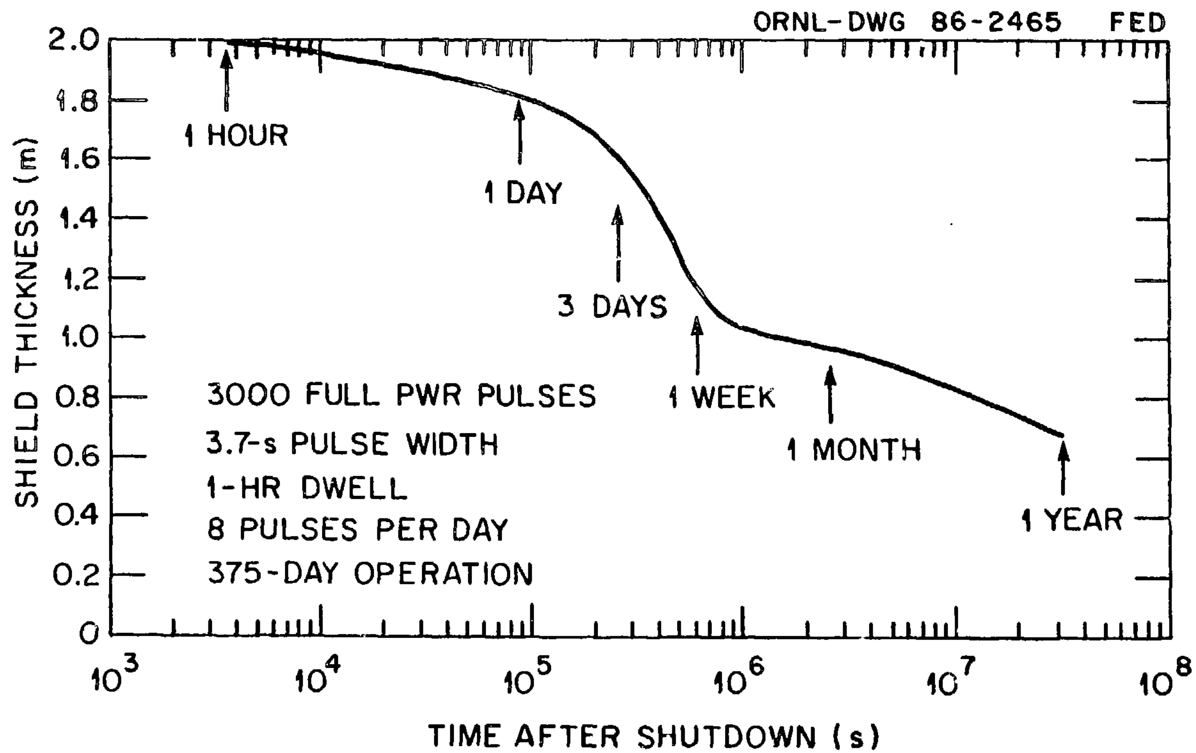
OPERATING SCENARIO

Design analysis corresponds to:

3000 full power pulses
3.7-second pulse width
1 hour cooling between pulses
8 pulses / day
375 day operation
24 hour post-shutdown cooling

Saturation of 24-hour dose rate occurs after 5 days of operation

Extending the cooling period reduces the shield thickness required to satisfy the 1 mrem/h criterion



MAJOR NEUTRONIC CONCERNS IN THE CIT

- **Radiation streaming from penetrations**
  - **vacuum duct port**
  - **rf ports**
  - **fueling system port**
  - **diagnostic ports**
  - **remote maintenance ports**

- **Radiation damage to reactor components**

- **Activation of test cell air atmosphere and coolant streams**
  - **safety assessment**

- **Activation of components at decommissioning**
  - **waste disposal characteristics**

Elevation view of the CIT showing major neutronics concerns

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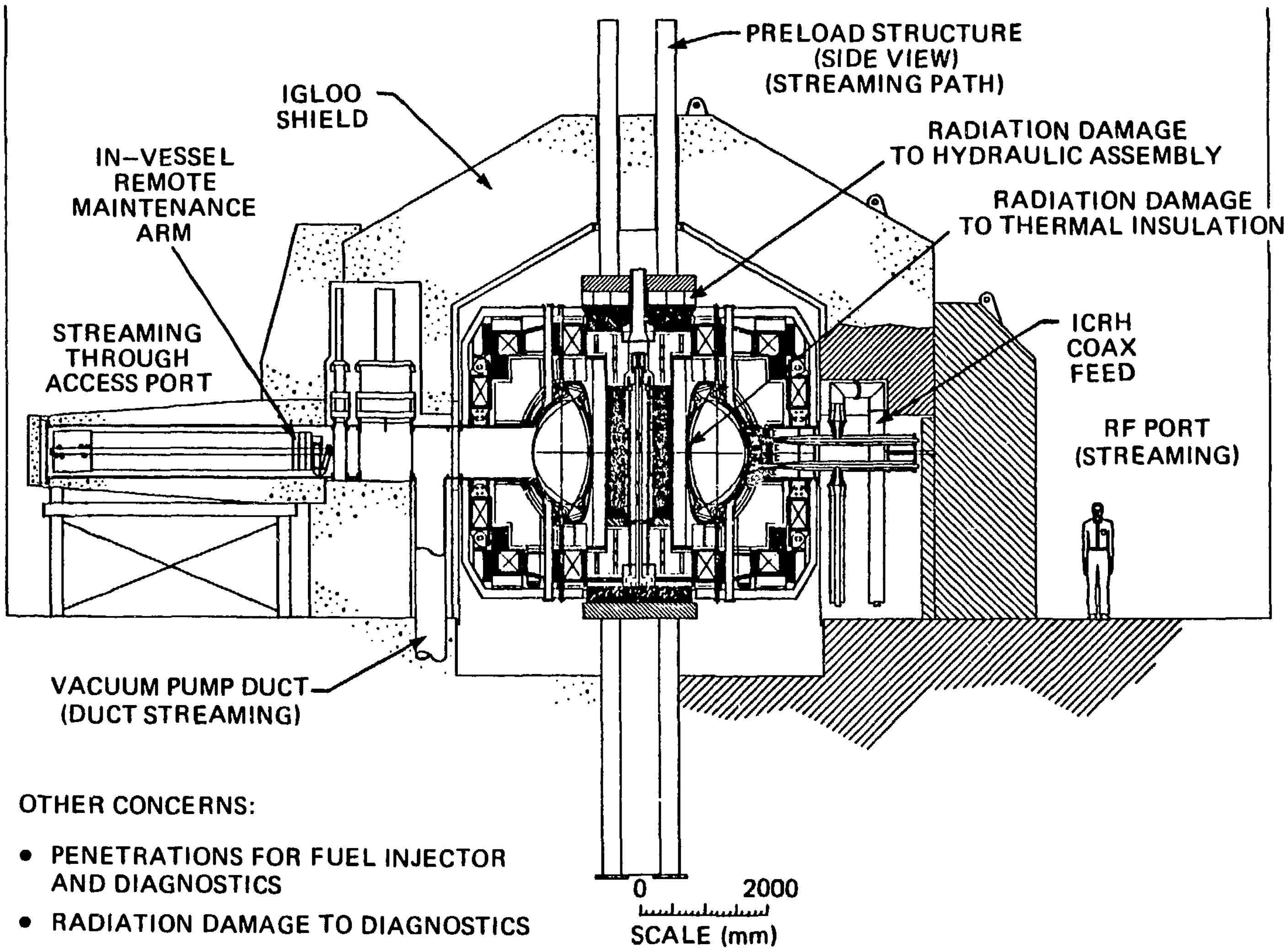

OTHER CONCERNS:

- PENETRATIONS FOR FUEL INJECTOR AND DIAGNOSTICS
- RADIATION DAMAGE TO DIAGNOSTICS

SUMMARY

- **Conceptual design of igloo shield has been completed**
  - **nominal thickness of 1.8 m based on 1 mrem/h criterion at 24 h after shutdown outside of bulk shield**

- **Advanced conceptual design work is in progress to evaluate major neutronics concerns:**
  - **radiation streaming**
  - **radiation damage**
  - **air and coolant activation**
  - **waste disposal characteristics**