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TMI-2 FUEL CANISTER INTERFACE REQUIREMENTS FOR INEL

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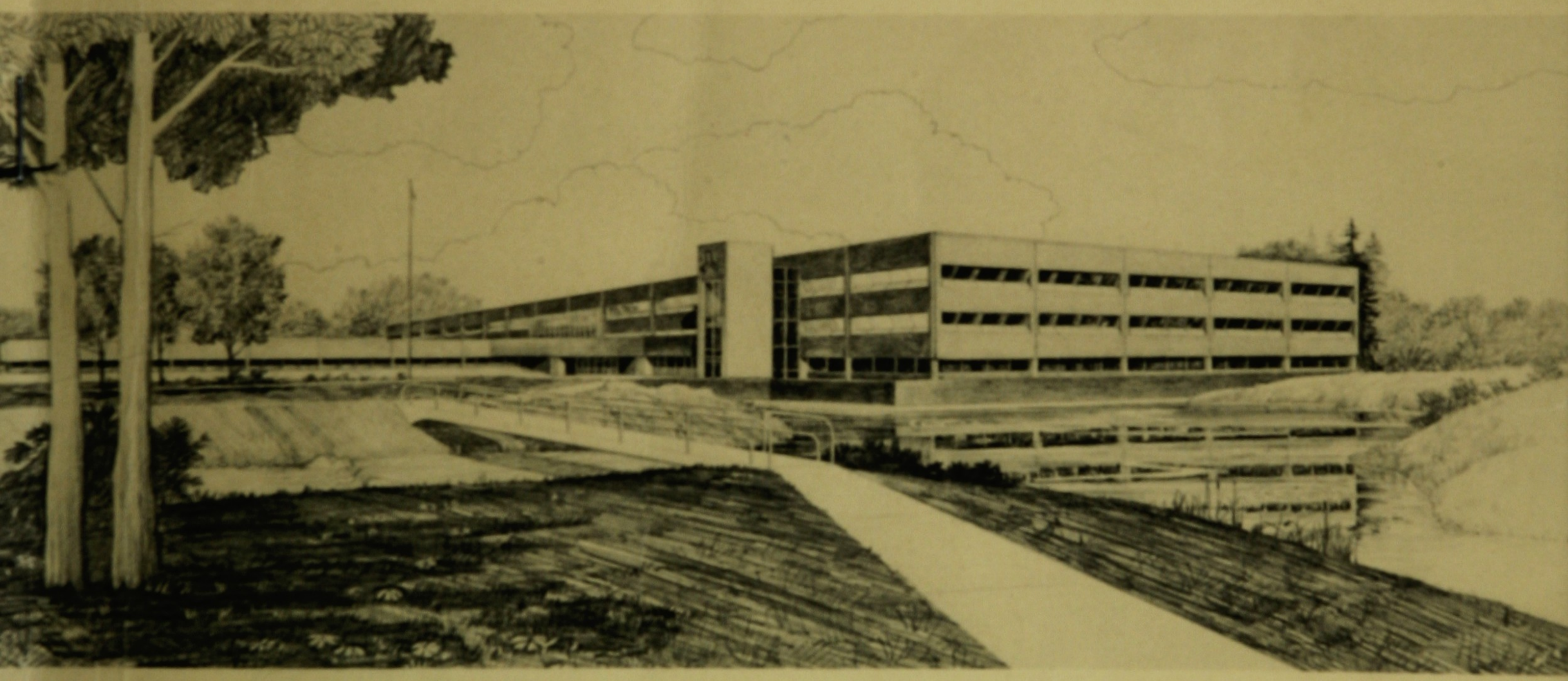
Idaho National Engineering Laboratory
Operated by the U.S. Department of Energy



This is an informal report intended for use as a preliminary or working document

Prepared for the
U.S. Department of Energy
Under DOE Contract No. DE-AC07-76ID01570

EG&G Idaho

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by
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TMI-2 FUEL CANISTER INTERFACE REQUIREMENTS FOR INEL

INTRODUCTION

Fuel assemblies and core debris from the damaged Unit-2 reactor of the Three Mile Island Nuclear Power Station (TMI-2) will be removed and transported to the Idaho National Engineering Laboratory (INEL) for interim storage, examination, and preparation for final disposal. Through the TMI-2 Core Examination Program, data will be gathered and analyzed. The information gained will provide a more complete understanding of the accident sequence which occurred in the TMI-2 reactor and will provide a better understanding of nuclear fuel behavior during other degraded cooling situations. Interim storage in the Water Pit of the Test Area North (TAN-607) Complex of INEL will support the examination program and facilitate preparation for final disposal at a selected repository.

The TMI-2 fuel will be loaded into stainless steel canisters and transported in licensed shipping casks to INEL. Upon receipt, fuel canisters will be unloaded in the TAN-607 Hot Shop and placed in the Water Pit for storage. The design concept for storage racks and unloading equipment at INEL assumes a maximum of 250 canisters. The design of the storage racks depends on the canister dimensions and must be coordinated with canister design.

This report focuses on fuel canister interface requirements at INEL which should be incorporated into the canister design criteria. The requirements will ensure compatibility with existing INEL structures and equipment to be used for receipt, unloading, and storage of fuel canisters. INEL can and does receive and store radioactive materials in many different forms, including reactor fuel. INEL requires detailed descriptions of canisters and casks. Therefore, requirements listed below represent engineering design features which will simplify the handling and storage operations; consequently, they are not to be viewed as absolute or non-negotiable. However, the core acquisition contract was negotiated with certain storage assumptions which effect costs of storage. Deviations from those assumptions which significantly effect costs would require approval by DOE-ID. If some stated requirements are too restrictive, modifications based on sound engineering principles may be negotiated with INEL.

Receipt, unloading, handling, and storing of fuel canisters at the TAN-607 Complex will conform with appropriate codes, standards, and regulations including:

DOE/ID-5440	Implementation of the National Environmental Policy Act
DOE/ID-5480.1	Chapter I--Environmental Protection, Safety and Health Protection for DOE Operations
	Chapter III--Safety Requirements for the Packaging of Fissile and other Radioactive Materials
	Chapter V--Safety of Nuclear Facilities
	Chapter XI--Requirements for Radiation Protection
	Chapter XII--Preventive, Control, and Abnormal of Environmental Pollution
ANSI/ANS 8.1-83	Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors
ANSI N16.5-75	Guide for Criticality Safety in the Storage of Fissionable Materials
ANSI N16.9-75	Validation of Calculational Methods for Nuclear Criticality Safety
10 CFR Part 50 Appendix B	Quality Assurance

10 CFR Part 72.73		Criteria for Nuclear Criticality Safety
EG&G Safety Manual		
	Section 9010	Administrative Requirements for Fissile Materials
	Section 9020	Fissile Material Control Areas
	Section 9030	FMCA Safety Analyses
	Section 9040	Criticality Safety Review and Approval
	Section 9070	Shipping and Receiving Fissile Materials (all section listed are attached as Appendix A)

INEL REQUIREMENTS FOR THE FUEL CANISTERS

1. Criticality

Fuel canisters loaded with enriched UO_2 fuel shall remain subcritical under all conceivable loading, handling, and storage situations. If neutron absorbers are used inside fuel canisters, GPU Nuclear shall demonstrate their continued effectiveness under transport and storage conditions. GPU Nuclear may provide EG&G Idaho with a criticality analysis of loading parameters which demonstrates subcriticality of storage of fuel canisters on 18 inch centers in water. A $K_{eff} \leq 0.95$ is required at INEL for the storage array. Two independent criticality safety analyses are required by EG&G Safety Policy 9030 (see Appendix A). They will consider the most reactive conditions which conceivably can occur, including misplacement of the fuel canisters in storage racks, dropping of loaded fuel canisters into an already loaded rack, reflection of neutrons in/by water, and canister flooding with non-borated water. Specific design requirements and considerations are listed in Appendix B.

Selection of a fuel canister with an inside diameter of less than 10 inches would simplify greatly the criticality safety requirements for transport and storage at INEL. Fuel canisters with an inside diameter greater than 10 inches will require internal neutron absorbers and/or geometrical constraints. If neutron poison materials or geometrical constraints are required inside or between fuel canisters in storage racks, their existence and location shall be verifiable periodically, and they shall remain in place during a credible accident or seismic event. Verifying their location and effectiveness will be a difficult and expensive task which can be avoided by choosing a fuel canister with an inside diameter of less than 10 inches which will have a K_{eff} <0.95 without neutron absorbers.

2. Maximum Envelope

The exterior wall of fuel canisters shall be straight ($\pm$ 0.5 inches in 14 ft), with a constant cross-section at locations where lateral support is provided by storage racks. The preliminary design concept of the storage racks envisions cylindrical fuel canisters with a maximum outside diameter of 13.375-inches located on 18-inch centers in the Water Pit. Smaller fuel canisters can be accommodated, but larger diameters would require larger spacing for the neutron absorbing material.

Projections are not acceptable which interfere with insertion or removal of a fuel canister from a storage rack. Chamfered or rounded edges on the bottom would make insertion and withdrawal easier. A surface roughness of less than 250 microinches (root mean square) is desirable.

3. Weight

The weight of a fuel canister plus its contents shall not exceed 2800 lbs total. That weight is compatible with facility floor loading constraints and will facilitate storage in the Water Pit on 18 inch centers. An empty canister must have negative buoyancy in water.

The maximum loading on the concrete floor of the Water Pit has been analyzed at 900 lb/ft^2 (excluding the water). If weight is transferred to structural girders under the floor, the maximum floor loading can be increased to 1800 lb/ft^2. The 1800 lb/ft^2 translates to 4050 lbs. maximum on each 18 x 18 inch area. That is equivalent to the weight of one canister, plus its share of the storage rack, plus a margin of safety. [At this time, a weight of the storage rack has not been established.]

4. Closure Head Design

The closure head shall include provisions for its remote removal and reinstallation while the fuel canister is in either a vertical or horizontal position. A bolt-on closure head with bolts that can be removed remotely by a lock-on wrench is preferred over a threaded closure. Seal welds should not be used unless required for shipping. If seal welds are used on fuel canisters, there should be a bolt-on head under a removable, seal-welded cover, particularly in those canisters identified for inspection/study by INEL. Fuel canisters which will not be opened at INEL may have welded closures only, provided the vent mechanisms are accessible. If the closure head of the fuel canister is a threaded type, pilot guides should be used to guide initial engagement of threads, thereby avoiding thread damage.

5. Canister Lifting Mechanisms

Lifting features shall be incorporated into the design of fuel canisters. The features may be placed either on the body near the top of the fuel canister, or on the closure head of the fuel canister.

A grappling mechanism that is self-aligning both horizontally and radially is essential for remote handling in the TAN-607 Hot Shop. The grapple should center itself over the canister closure head, even if the lifting crane is slightly off center. It also should rotate automatically, permitting the lifting devices to engage properly and positively. The lifting mechanism should be designed with a structural safety factor of at least five times the weight of a loaded fuel canister.

6. External/Internal Pressure

The canister shall be designed to withstand a minimum external pressure of 30 psig for storage in the Water Pit of TAN-607. Requirements for minimum internal pressure related to buildup of radiolytic gases are the function of transportation. Those requirements are acceptable to INEL.

7. Vertical Drop

The fuel canister shall withstand (not rupture or break) a 10-foot drop in air onto a concrete surface when fully loaded. Some deformation is acceptable, provided it does not interfere with lifting by the grappling mechanism. During unloading of the cask in the TAN-607 Hot Shop, the fuel canister will be lifted approximately 20 ft vertically, then lowered to a height below 10 ft before being moved it to a storage rack in the Vestibule of the Water Pit.

8. Materials

The design life of all components of the canister shall be $\geq$30 years. All materials used in construction of the fuel canister shall be compatible with the chemistry of the Water Pit. Principal chemical characteristics of the Water Pit are as follows:

- Cl^- < 10 ppm
- F^- < 1 ppm
- Suspended solids < 100 ppm
- Conductivity < 60 micromhos/cm
- pH 4.2 - 10.5.

Aluminum and aluminum alloys shall not be used for any part of the fuel canister which would be exposed to water. Materials used in the fuel canister, including the gasket seals, must withstand a cumulative radiation exposure of 2×10^8 rads (gamma). If the fuel is located only in the bottom end of a canister, the cumulative radiation exposure to gaskets will be reduced to less than 500 rads (see Appendix C) because of shielding characteristics of four feet of water inside the canister above the fuel material. That may prevent a relaxation of the gasket exposure criterion. Cladding of neutron absorbers by corrosion-resistant materials is mandatory.

9. Vents and Drains

Vents and/or drains shall be re-usable and shall be located where they are protected from the grapple lifting device. Recessing vents and/or drains into the closure head or the closure head support would protect them from damage in ordinary situations. A quick disconnect with a positive closure is required. The type recommended by design engineers of EG&G Idaho is a 1-inch stainless steel quick disconnect valve (models LL8-H36/LL8-K36), manufactured by The Hanson Company of Cleveland, Ohio.

10. Identification

A unique serial number shall be stamped on the head of each fuel canister in characters at least 1 inch high. A bill of lading indicating serial number, gross weight, net weight, and contents shall accompany each fuel canister.

11. External Contamination

Loose external contamination shall be less than 10,000 dpm beta/100 cm^2 and 250 dpm alpha/100 cm^2. Design of the exterior of the fuel canister should facilitate easy decontamination of all sides. The recessed vents and bolts should have sufficient clearance to permit cleaning with special brushes.

12. Quality Assurance

A quality assurance program shall be prepared in accordance with 10 CFR 50, Appendix B.

13. Instruction Manuals

Manuals shall be provided which clearly indicate recommended methods for (a) opening and closing the fuel canister, and (b) venting, lifting, and supporting the fuel canister.

FACILITY DESCRIPTION

The TAN-607 Complex of INEL will be used for receipt and interim storage of fuel canisters, and for examination and preparation of the core debris for final disposal. Figure 1 illustrates the TAN-607 Complex and identifies the Hot Shop, Warm Shop, Water Pit, and TAN Hot Cell. Descriptions of those specific areas and equipment are presented below for general information.

Hot Shop

The Hot Shop is a shielded facility designed for service and maintenance of large, experimental nuclear assemblies using remote handling equipment. The shop is 51 ft wide, 160 ft long, and 55 ft high, with concrete walls 7 ft thick. Nine shielded viewing windows link the operating galleries to the Hot Shop as shown in Figure 2. The main door to the Hot Shop (located on the west end) is 28 ft wide and 32 ft high. The allowable floor loading in the Hot Shop is 250 lb/ft^2; however, concentrated loads can be accommodated in specific areas.

Overhead Bridge Crane

The overhead bridge crane is rated at 100 tons and ranges over the entire shop, except for a 12-foot section next to the main doors. The overhead crane also contains an auxiliary 10-ton hoist on the same trolley. The service height for the crane, as well as other equipment, is shown in Figure 3.

Overhead Manipulator

The bridge mounted, overhead (electromechanical) manipulator is capable of manipulating large tools and hardware. It services the entire shop to a height of 30 feet. The manipulator hand can lift 500 lb, and the shoulder hook lifts 5000.

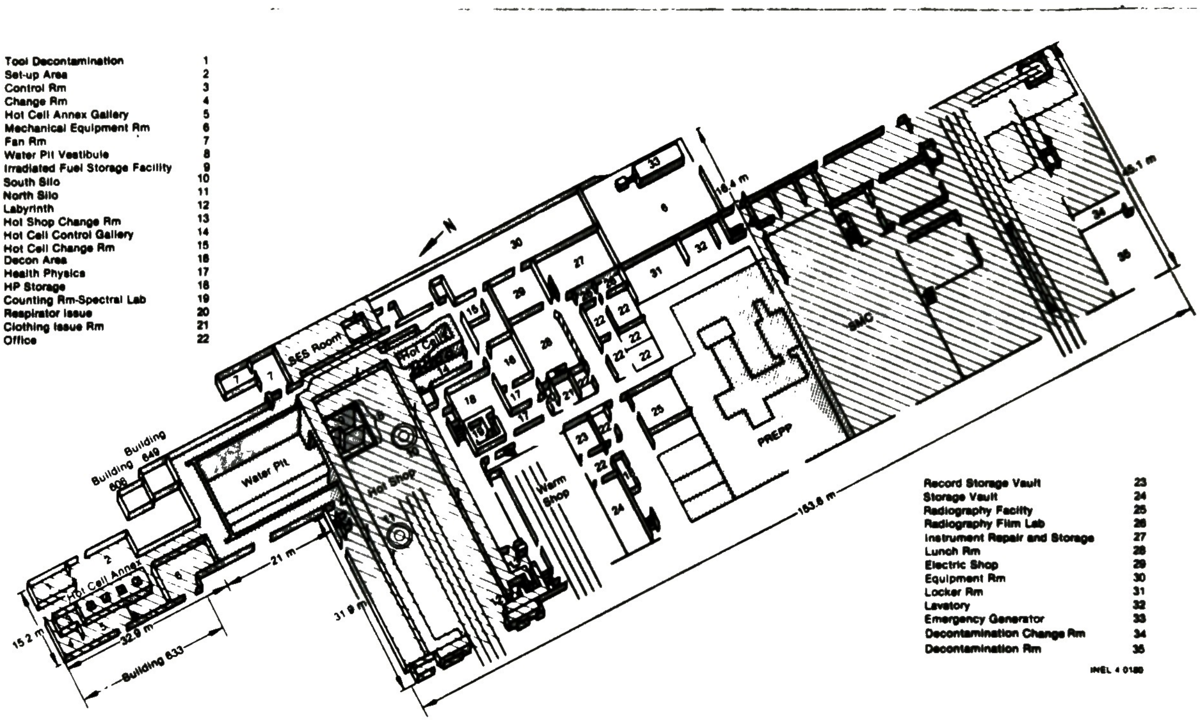


Figure 1. Diagrammatic representation of the main floor of the TAN-607 Complex.

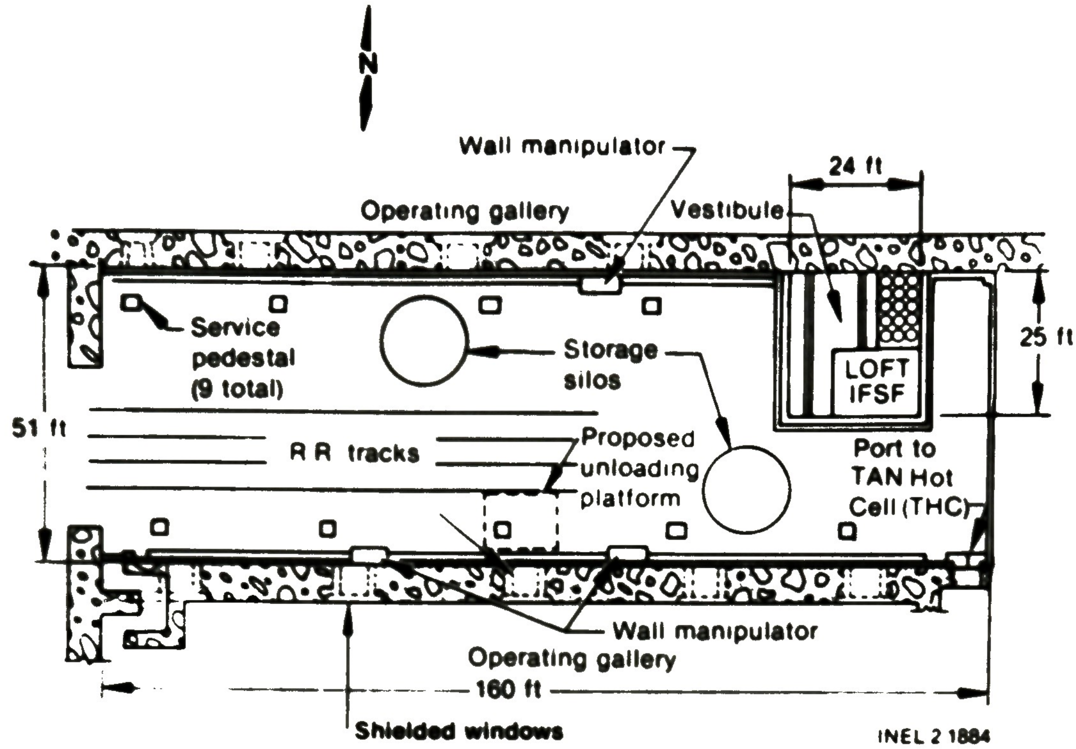


Figure 2. Plan of Hot Shop in TAN-607.

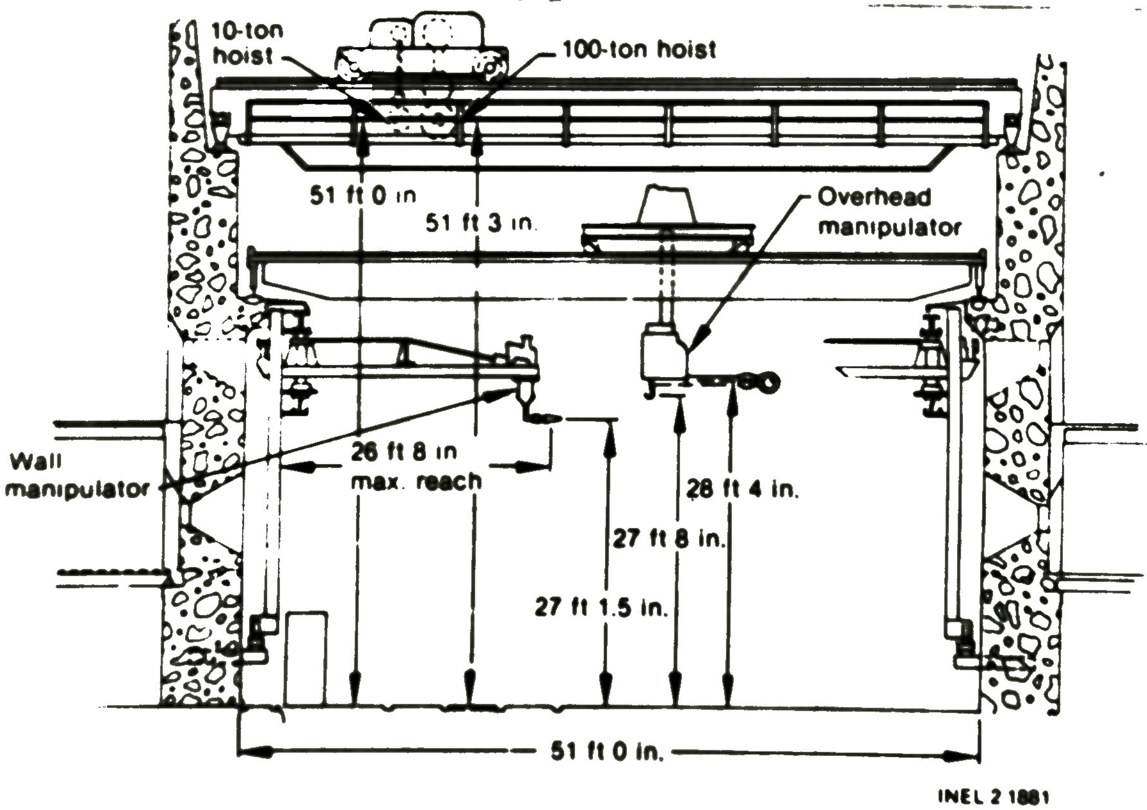


Figure 3. Elevations of remote handling equipment in the TAN-607 Hot Shop.

Manipulators

Wall-mounted manipulators (one on the north wall and two on the south wall) can travel both horizontally and vertically along the Hot Shop walls; however, their horizontal mobilities are limited by storage silos shown in Figure 1. Jib booms of the wall-mounted manipulator(s) can be swung from the wall to the center of the shop. Figure 3 presents an elevation view of the equipment. Window G, at the northwest corner of the Hot Shop, includes a pair of heavy-duty master-slave manipulators that can be used to maneuver objects at that window.

TAN Hot Cell

The TAN Hot Cell adjoins the southeast end of the Hot Shop as shown in Figure 1. It is designed for observation, disassembly, and examination of radioactive components. As shown in Figure 4, the Hot Cell is 35 ft long and 10 ft wide. Four viewing windows, each equipped with master-slave manipulators, allow personnel to work remotely from the operating gallery. Two bridge-mounted, overhead manipulators service the entire Hot Cell. Each manipulator has a 150 lb hand capacity, 750 lb shoulder hook capacity, and a 2-ton chain hoist. An elevation view of the Hot Cell is shown in Figure 5.

Access to the Hot Cell from the Hot Shop is provided through two shielded sliding doors covering an opening 46 inches wide and 8 ft high. Components are transported into the Hot Cell on a remotely-operating air pallet, which has a 2-ton capacity. The false floor inside the Hot Cell is capable of supporting a 2-ton load.

Water Pit

A Water Pit is located adjacent to the north side of the Hot Shop (see Figure 1). The concrete-walled pit is 48 ft wide, 70 ft long, and 24 ft deep, as shown in Figures 6 and 7. Access to the Water Pit is through the Vestibule located in the northeast part of the Hot Shop. Fuel is lowered

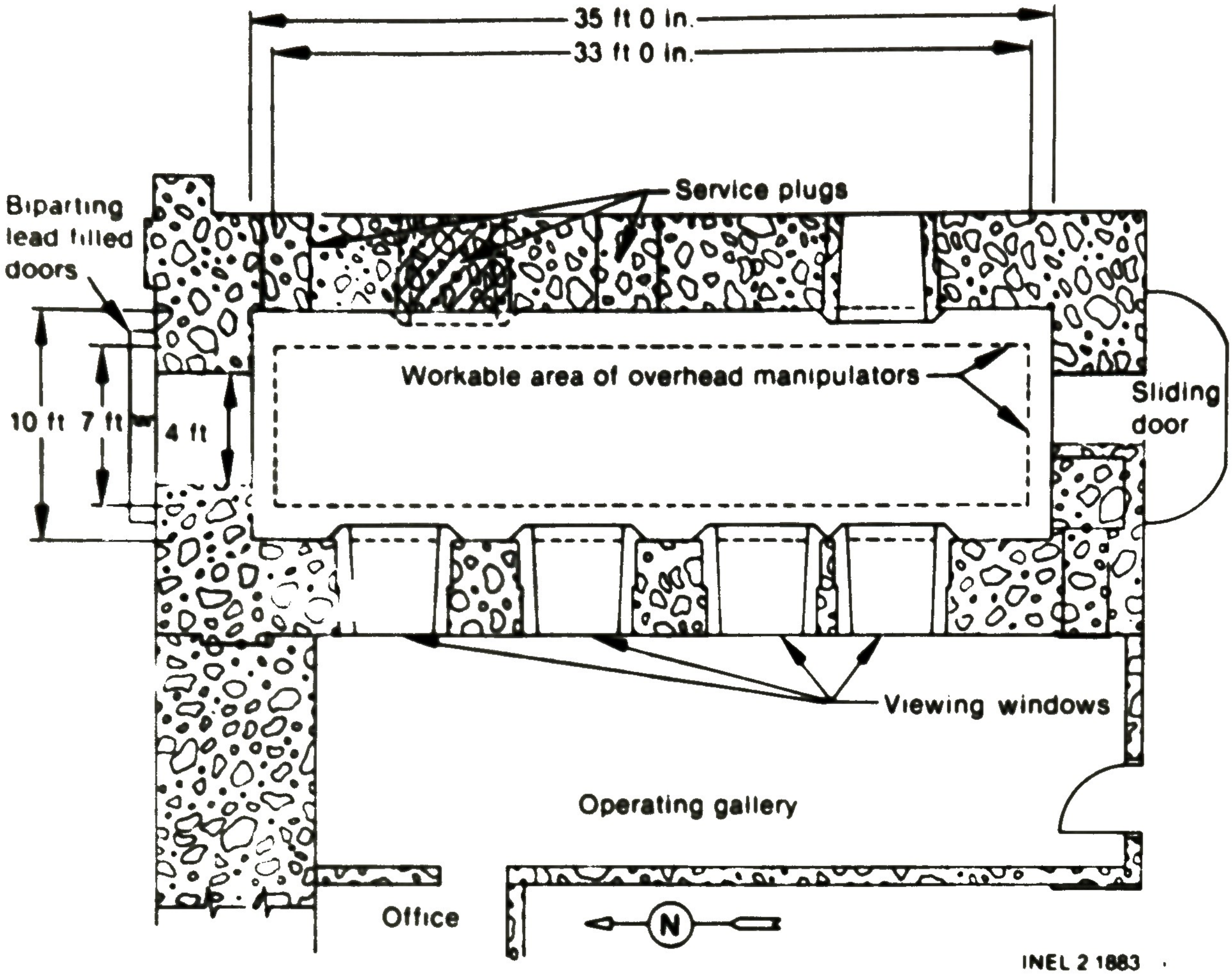


Figure 4. Plan of TAN Hot Cell of the TAN-607 Hot Shop.

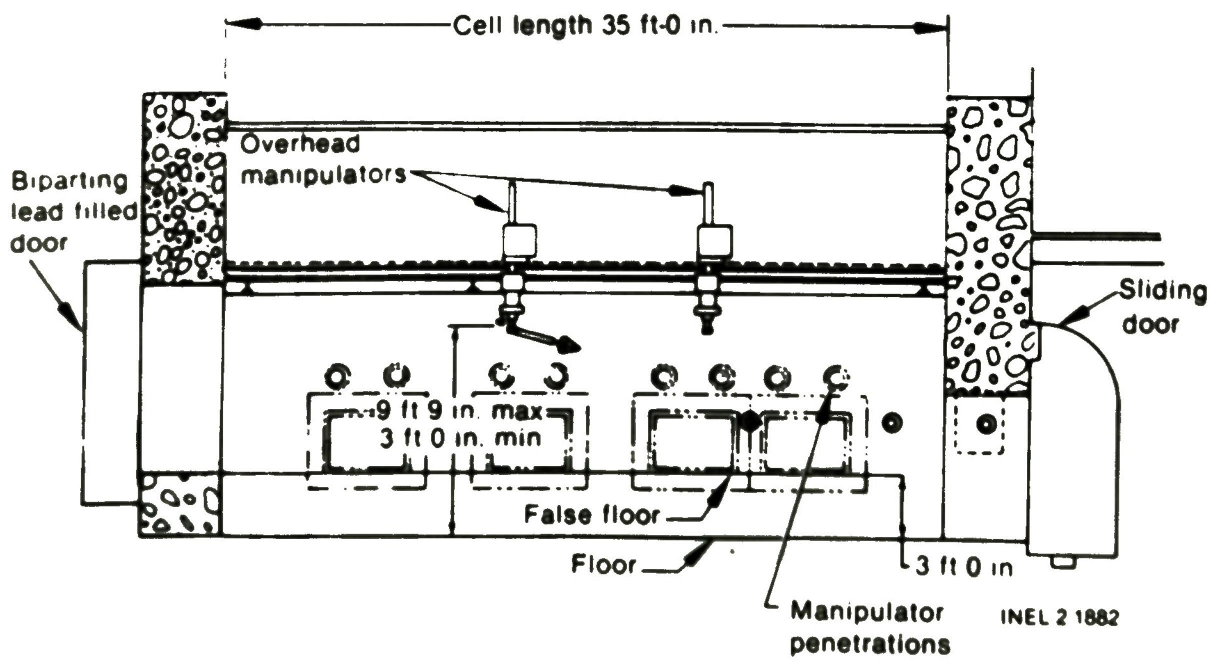


Figure 5. Elevation view of TAN Hot Cell, showing handling equipment.

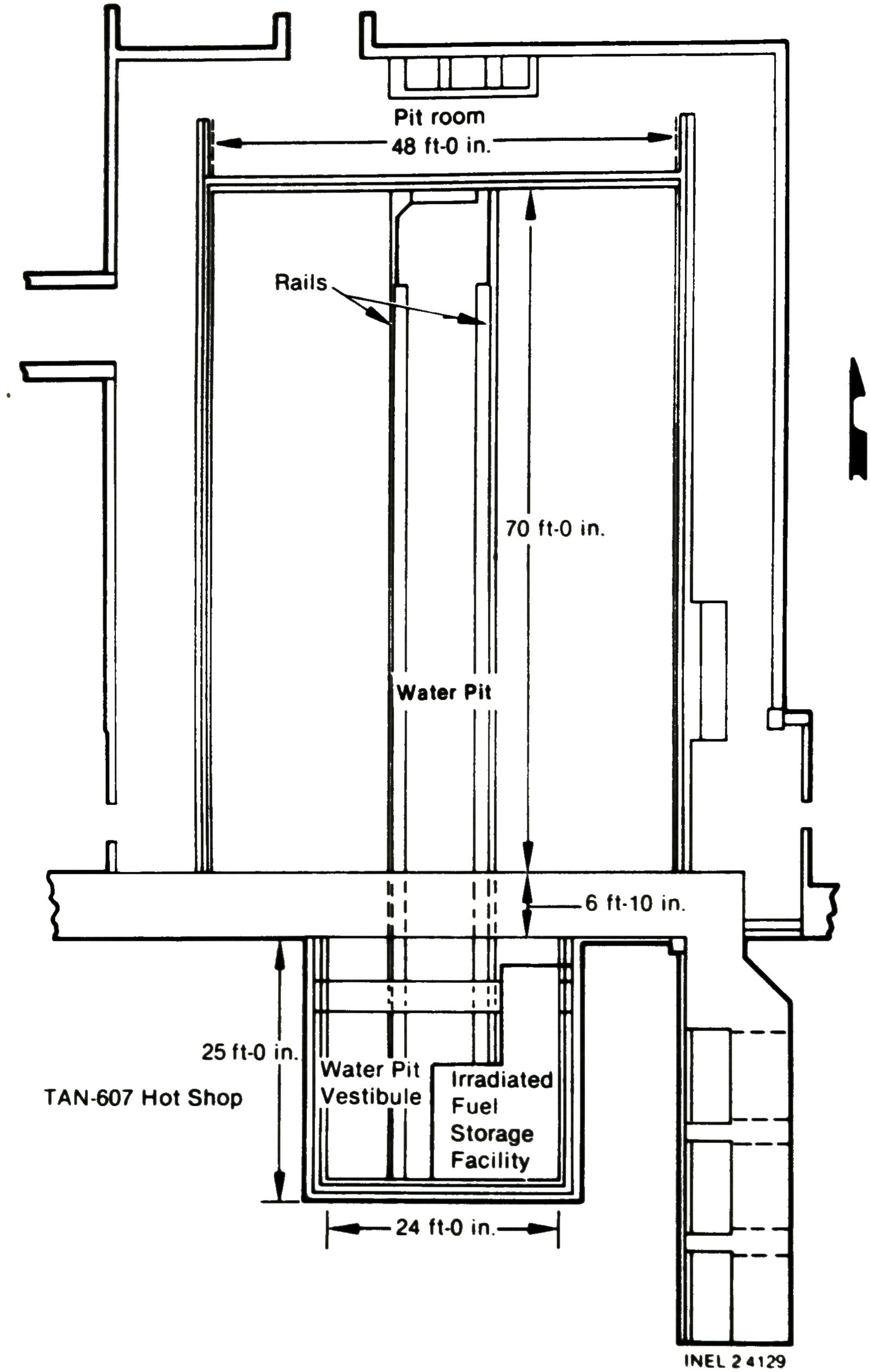


Figure 6. Plan of Water Pit adjacent to the TAN-607 Hot Shop.

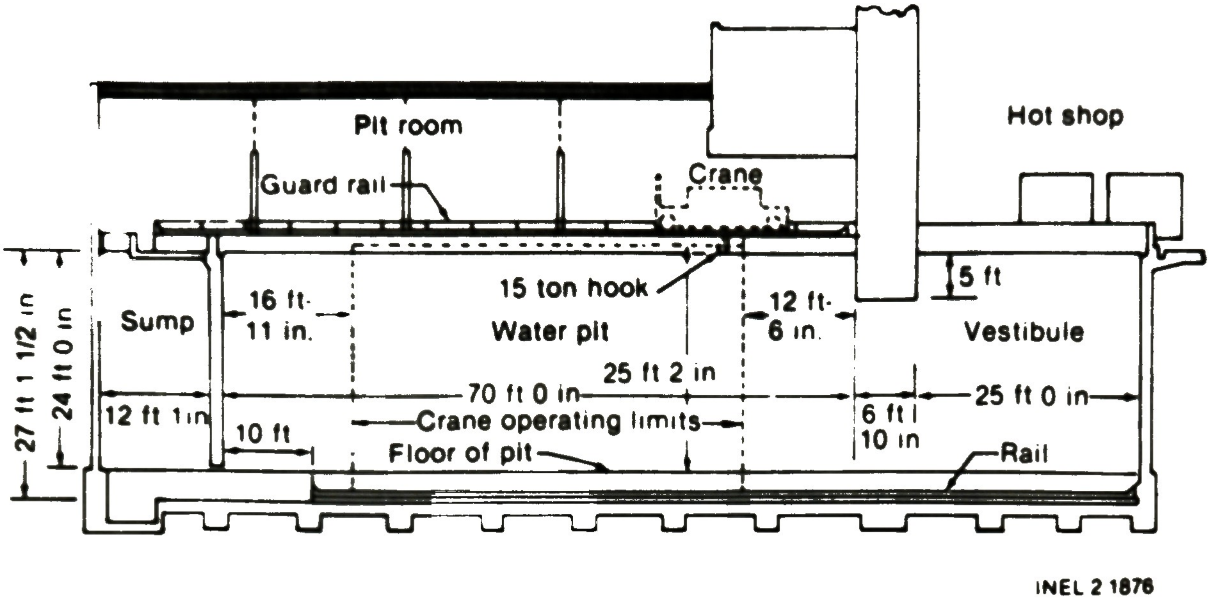


Figure 7. Elevations of Water Pit and Vestibule.

into the Vestibule and transferred to the Water Pit via a cart which traverses the bottom-middle of the Water Pit. The cart, shown in Figure 8, is moved by a power winch located on the north end of the Water Pit. A 15-ton bridge crane services the entire Water Pit, except for a 9-12 ft strip at either end; the crane also serves as a work platform over the Water Pit for manipulating pool tools.

Warm Shop

The Warm Shop, located to the south of the Hot Shop, is a service area for handling test assemblies with low levels of radioactive contamination. The area is 51 ft wide, 80 ft long, and ~50 ft high. It is serviced by an overhead bridge crane with 30/5-ton capacities. Travel limits of the bridge crane are as shown in Figure 9. The Warm Shop has a floor drain system connected to the hot waste holding tank of TAN-607 Complex.

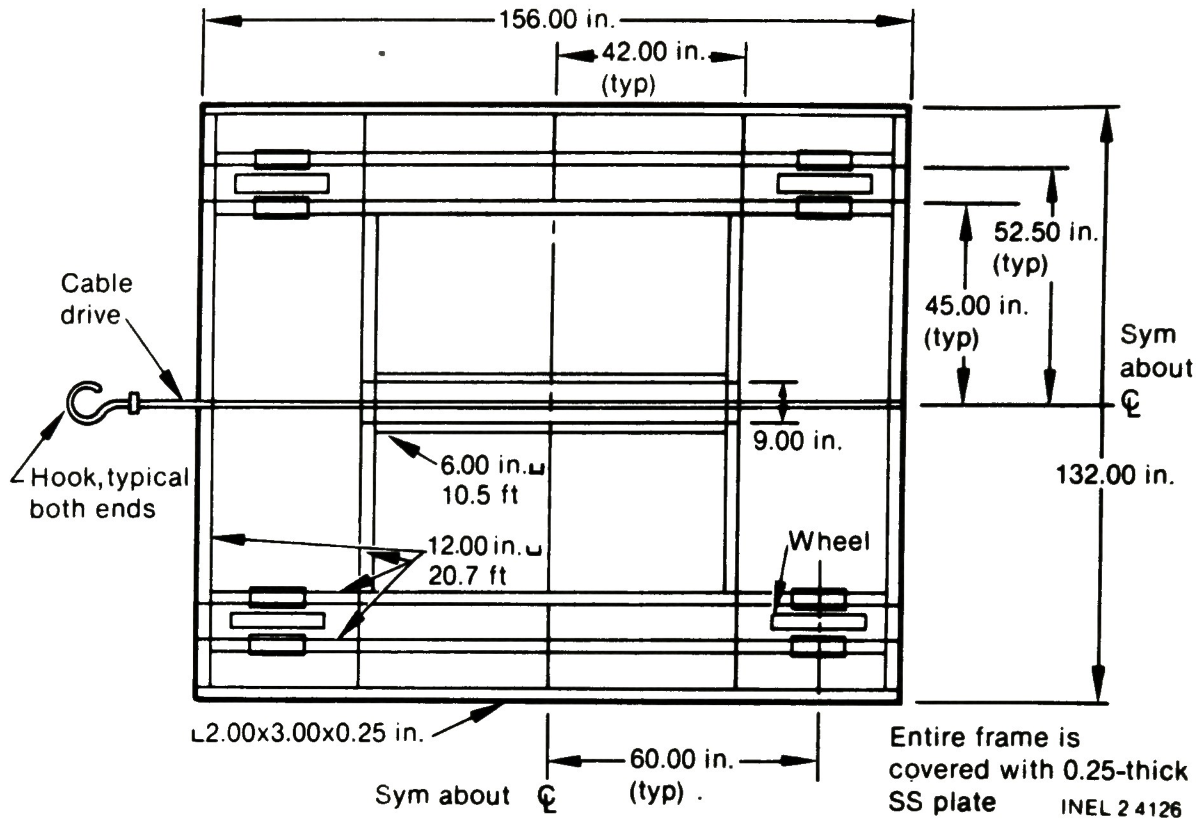


Figure 8. Old schematic of cart in the Water Pit.

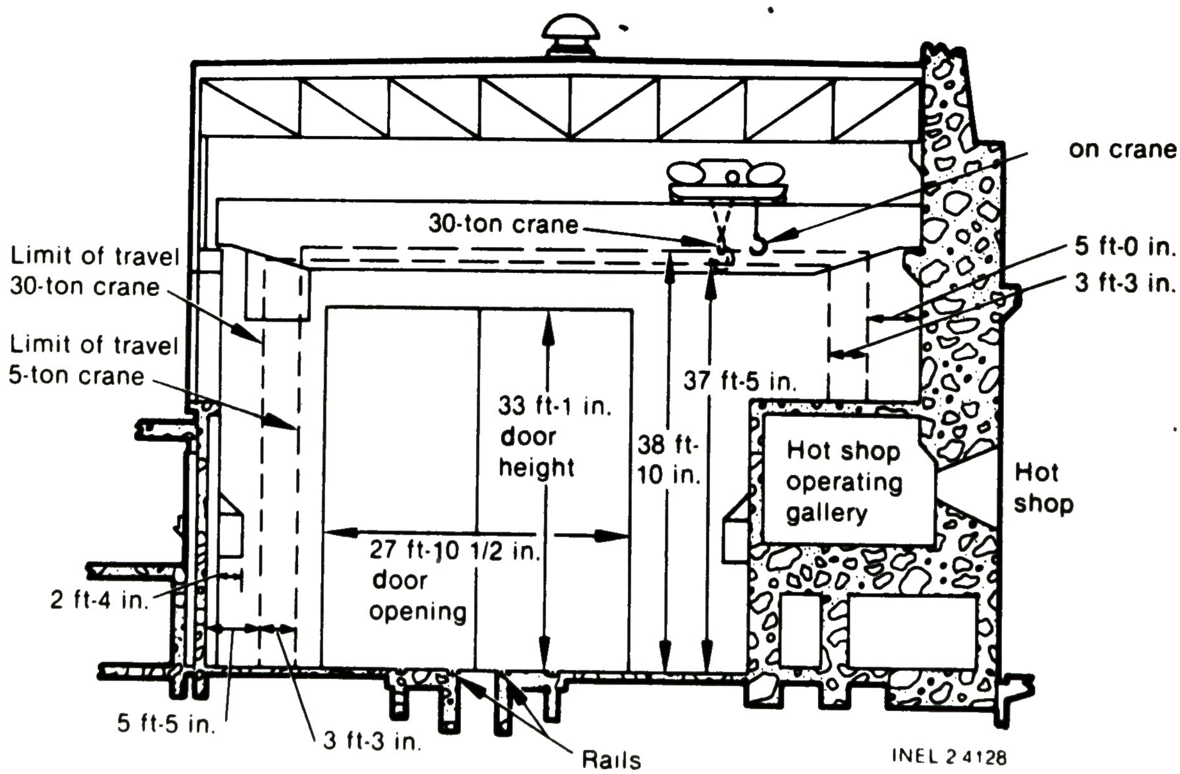


Figure 9. Elevation of the Warm Shop showing working limits and distances of the overhead bridge crane.

PROCESS DESCRIPTION

A process flow diagram (Figure 10) identifies interface requirements of the fuel canister within the TAN-607 Complex. The assumptions used the flow diagram are:

- The damaged TMI-2 fuel will be loaded into fuel canisters, placed in either a legal-weight truck casks or in a rail cask, and shipped to INEL
- Each fuel canister will be drained, purged with an inert gas, and pressurized to about two atmospheres of argon
- Each fuel canister will contain partially disintegrated fuel assemblies or pulverized with fuel other materials
- The gross weight and net contents will be identified and indicated
- Fuel canisters will have negative buoyancy in water.

Receipt and Unloading of the Shipping Cask

Legal-weight truck casks will be received at the TAN security gate, escorted to the TAN-607 Complex, and driven inside the Warm Shop. [In the event rail casks are used, the rail cask will be transferred to a heavy duty transporter at the Central Facilities Area and escorted to the TAN-607 complex.] Inside the Warm Shop, the cask and trailer will be de-iced (if required) and surveyed for radioactive contamination. The tiedowns will be removed, and the existing 30/5-ton crane will be used to remove any shipping devices (i.e., impact limiters). The trailer and shipping cask then will be moved from the Warm Shop to the TAN-607 Hot Shop. A lifting yoke will be attached to the cask and the 100-ton crane will be used to orient the cask in a position where the closure devices can be removed. A steel working platform will be designed and constructed to permit easy access to the shipping cask.

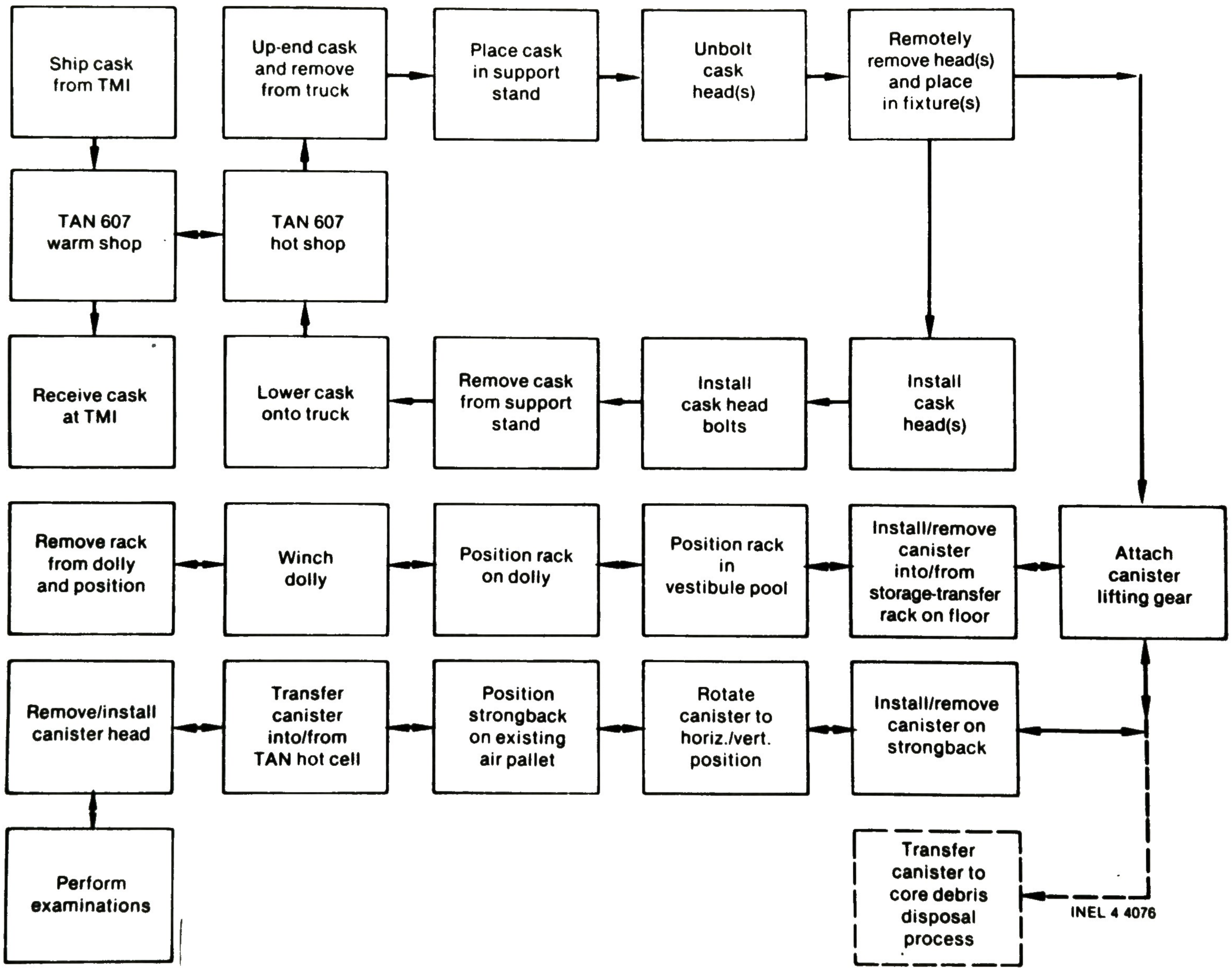


Figure 10. INEL canister-handling flow diagram.

Depending on the type of shipping cask, procedures will be developed for remotely removing the inner-most closure head or heads, providing access to the fuel canister inside. Using the 10-ton bridge crane and canister lifting grapple, the fuel canister will be removed from the shipping cask and transported to a storage rack located in the Vestibule of the Water Pit. The self-aligning feature of the canister lifting grapple is extremely important because the remotely handled lifting grapple will be operated blindly as it is lowered into the shipping cask for attachment to the fuel canister. If venting is required, the venting connections will be made to the fuel canister at this time. Water in the Vestibule will provide shielding of the fuel canister, allowing the empty cask to be inspected, reassembled, and attached to the transporter for the return trip to TMI. When the storage rack is loaded fully with fuel canisters, it will be moved into the Water Pit and mechanically coupled to other full fuel racks. The storage racks are assembled to form a storage rack capable of withstanding seismic accelerations of up to ± 1.7 g horizontally and ± 0.80 g vertically.

To retrieve a specific fuel canister from the Water Pit, any intervening racks (three maximum) will be disconnected temporarily and relocated to a side position in the Water Pit. The selected rack with the specific fuel canister will be placed on the transfer cart and moved to the Vestibule where the fuel canister will be withdrawn and placed in the strongback (Figure 11). This will be accomplished with the canister-lifting grapple tool attached to the 10-ton crane. The procedure for rotating the fuel canister to a horizontal position in the stongback is described in Appendix D.

Using the existing air pallet system, the strongback assembly and fuel canister will be moved to the TAN Hot Cell. Once inside the Hot Cell, the strongback assembly will be removed from the transporter and the transporter will be moved outside the Hot Cell. With the canister in a horizontal position, the canister closure head will be removed using remote handling devices. The closure head must be designed to make removal remote handling devices possible while the canister is in a horizontal position.

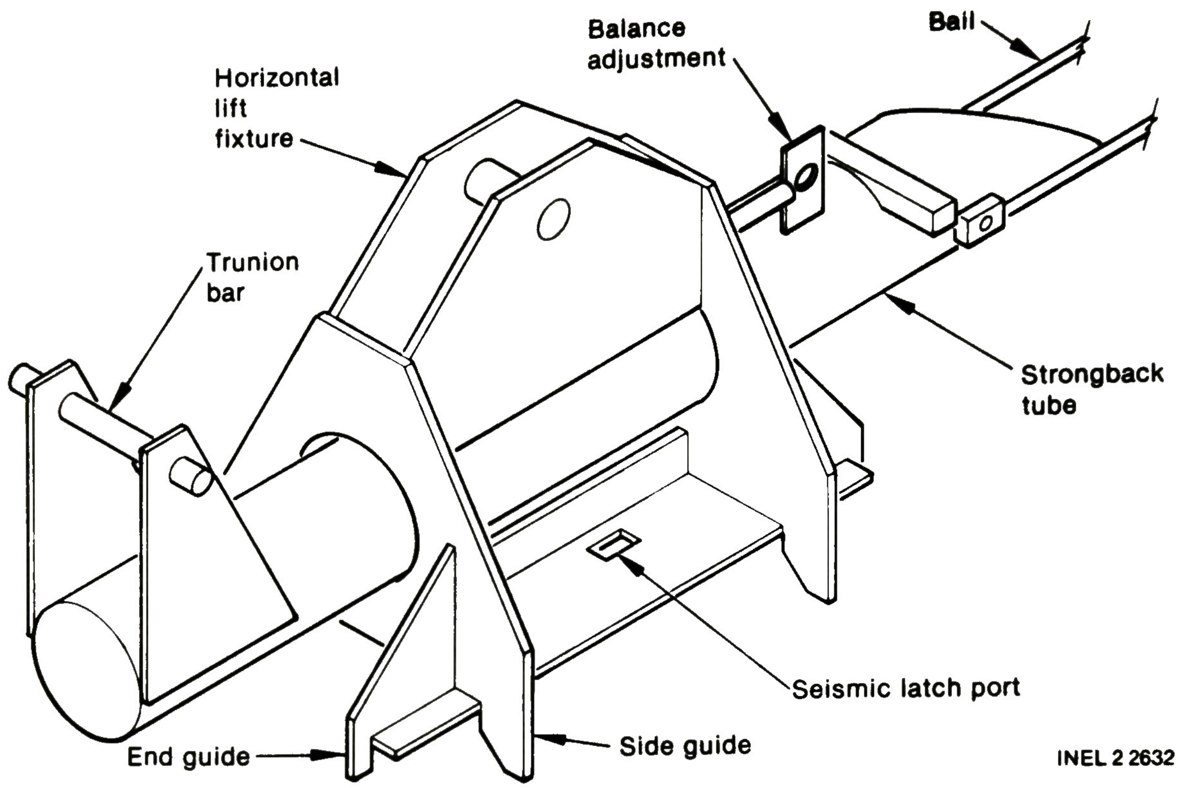


Figure 11. Strongback Assembly.

The vent and drain connections must be reusable for recharging the fuel canister with an inert gas after the examination is completed and the closure head reinstalled. The fuel canister will be returned to the Water Pit via reversing the procedure just described.

FUEL STORAGE RACKS

Fuel storage in the Water Pit will be in storage racks, each holding as many as six fuel canisters. The spacing of fuel canisters will be 18-inch center-to-center. It is anticipated that external neutron absorbers will be required to provide a safe subcritical configuration with the 18-inch spacing. The need for external neutron absorbers in the event of leakage of fuel canisters, water loss from the Water Pit, or accident or seismic disturbance will be evaluated when the fuel canister design and loading parameters are established.

The maximum number of fuel canisters which can be placed in the TAN-607 Water Pit on 18-inch centers is 400. However, that would use all of the available storage space in the Water Pit area, and preclude programmed and other planned uses of the Water Pit. To support the proposed defueling and shipping schedule, technical specification have been prepared for a modular storage rack system containing a maximum of 256 canisters. Calculations indicate that the temperature of the Water Pit will not exceed 70°F with the full TMI-2 core in storage. It is anticipated that convective and evaporative heat losses will keep the water temperatures well below 70°F.

APPENDIX A

SELECTED SECTIONS FROM SAFETY MANUAL OF EG&G IDAHO

EG&G Idaho Inc

SAFETY MANUAL

Title: ADMINISTRATIVE REQUIREMENTS FOR FISSILE MATERIALS

Approved

Legend
*Revision
#Addition

(This revision involves updating organization titles and minor editorial changes only)

This document establishes administrative requirements and responsibilities for criticality safety in the handling, processing, shipping, receiving, and storing of fissile materials (defined in SM-9020) in EG&G-controlled activities outside of reactor cores.

* SAFETY RESPONSIBILITIES

* HEALTH AND SAFETY DIVISION shall:

1. Establish safety standards for all operations involving fissile material outside of reactor cores.

2. Conduct periodic reviews, appraisals, and surveillance to ensure EG&G compliance with applicable codes, standards, and regulations.

3. Provide technical assistance in meeting the requirements of criticality safety.

4. Review and approve all fissile material-related control procedures and safety analyses, in accordance with SM-9040.

COGNIZANT LINE MANAGERS shall:

1. Designate and post areas in which more than 15 g of fissile material may be handled, processed, or stored routinely as Fissile Material Control Areas (FMCAs), as required in SM-9020.

2. Ensure that a separate safety analysis is prepared (in accordance with SM-9030) for each FMCA in which more than 200 g of fissile material (400 g if only uranium-235 is involved) is to be handled, processed, or stored.

3. Develop formal control procedures for each FMCA, as required by SM-9020.

4. Obtain an independent review and approval of fissile material-related control procedures and safety analyses, as required in SM-9040.

5. Obtain, install, and maintain a criticality alarm system in FMCAs when required (see SM-9020).

6. Ship and receive quantities of fissile material in excess of 15 g, in accordance with SM-5070, SM-5080, and SM-9070.

7. Designate custodians and Fissile Material Handlers, and maintain personnel records in accordance with SM-9020.

8. Establish a criticality safety training program, in accordance with SM-4030.

REFERENCES

* 1. DOE Order 5480.1A/ID Order 5480.1, Chapter V, Safety of Nuclear Facilities.

2. Safety Manual 1050, Safety Appraisals.

3. Safety Manual 4030, Safety Qualification Courses.

4. Safety Manual 9020, Fissile Material Control Areas.

5. Safety Manual 9030, Criticality Safety Evaluations.

6. Safety Manual 9040, Criticality Safety Review and Approval.

7. Safety Manual 9070, Shipping and Receiving Fissile Materials.

EG&G Idaho, Inc SAFETY MANUAL	Title: FISSILE MATERIAL CONTROL AREAS	
	Approved:	Legend *Revision #Addition

(This revision involves updating organization titles and minor editorial changes only)

This document establishes responsibilities and procedures for identifying and operating Fissile Material Control Areas (FMCAs).

Fissile material is defined as uranium-233, uranium-235, plutonium-238, plutonium-239, plutonium-241, neptunium-237, americium-241, and curium-244. (Natural or depleted uranium [enrichment ≤ 0.75%] is not considered fissile material.)

NOTE: For criticality safety purposes, fissile material content for materials which have experienced burnup or irradiation in a reactor, the "as fabricated" or pre-irradiation isotopic content shall be used unless the fissile material content increases with irradiation, e.g., irradiating a breeder blanket (^{238}U) producing ^{239}Pu.

EXEMPTIONS

The following categories of fissile material are exempt from the requirements for handling, processing, or storing within an FMCA, under the conditions specified:

- Fissile material in amounts less than 15 g is exempt from the requirements for criticality safety and FMCA procedures.

- Radioactive sources used in radiography, reactor startup, counting room, and health physics activities may be used, handled, or stored outside FMCAs without specific criticality safety procedures, provided inventory control is sufficient to preclude the accumulations of quantities greater than 200 g of fissile material (400 g if only uranium-235 is involved) in any one location.

- A single ATR, ETR, or PBF reactor fuel assembly, fuel element, plate, or rod containing fissile material may be temporarily stored without a specifically approved procedure, provided a temporary FMCA is established by meeting the posting requirements of this section. Such storage requires oral approval daily from the Health and Safety Division.

- DOT- or DOE-approved shipping containers can be stored temporarily outside FMCAs, provided the sum of their transport indexes do not exceed 50 and the containers (as a group) are secured and separated by 4 m in air from other fissile material.

The areas at the Radioactive Waste Management Complex (RWMC) utilized for the receipt, receipt inspection, and the disposal or storage of radioactive waste materials are not to be considered FMCAs except for the following:

- FMCAs must be established for the storage of plutonium standards used for calibration.

- A Safety Analysis, per SM-9040, is required.

- The paragraph "Personnel" of this section is applicable to the RWMC. The safety limits will be stated in RWMC-controlled documentation.

- Waste packages, suspected of exceeding approved fissile limits, may be temporarily stored subject to the following:
  - The waste package container must be DOE approved
  - The waste package must be separated by 4 m in air from other fissile material
  - The storage area must consist of a rope barrier and FMCA posting per paragraph "Posting" below as a minimum.

REVIEWS AND APPROVALS

Reviews and approvals of FMCA Safety Analyses and control procedures shall be obtained, in accordance with SM-9040.

SECURITY

Security for the FMCA shall be established, in accordance with the DOE Orders 5630 series.

PROCEDURES

Except for operations specifically exempted above, all quantities of fissile material greater than 15 g shall be handled, processed, or stored in areas designated as FMCAs, in accordance with formal procedural controls. The FMCA shall be given a title distinct from other FMCAs. Controls shall be developed in accordance with the requirements below.

A Safety Analysis for each FMCA shall be developed by line management, in accordance with SM-9030 and SM-2040.

For each area, criticality safety control procedures (e.g., Standard Practices) shall be developed and be readily available to all personnel authorized to handle, process, or store fissile material in that area. The following items shall be used in the development of these procedures:

Posting

Vaults, dry storage facilities, work areas, and canals used for the handling or storage of quantities of fissile material greater than 15 g shall be posted as Fissile Material Control Areas. Signs used for posting FMCAs shall conform with the requirements of Appendix A of this section.

"No Water" FMCAs

If possible, all FMCAs should be established such that criticality safety is not dependent upon the exclusion of water. (If this is not possible, the Health and Safety Division shall prescribe special provisions.)

Personnel

Individuals whose work assignment includes the storing, handling, or processing of fissile materials or who are charged with its physical or administrative custody shall be approved by their supervisor as FMHs (Fissile Material Handlers). The training and qualification requirements are included in SM-4030. For each FMCA, a qualified FMH shall be assigned as custodian. Other FMHs may be assigned alternate custodians. The custodian and alternate custodian are charged with the administrative and physical custody of the

fissile material under their jurisdiction. Notification of such
* assignments or changes in assignments must be given to Criticality Safety, Health and Safety Division.

Safety Limits

Storage, handling, and processing limits shall be specified in formal procedures, usually in the form of EG&G Standard Practices or similar manuals. These limits shall be conspicuously posted within all FMCAs used as dry storage facilities unless shown to be unnecessary in the Safety Analysis. Safety limits and procedures for reactor main floors and canal storage shall be immediately available to all FMHs. All safety limits shall include:

- Storage and Processing Limits--overall FMCA limits expressed in isotopic mass or in numbers and description of units as well as the quantity and type of fissile material which may be in each storage or processing unit.

- Handling Limits--the total quantity and type of fissile material which may be out of approved storage or processing configurations. (The maximum amount of fissile materials which may be outside of approved storage or processing positions is also the maximum amount that may be transferred at one time from shipping containers.)

Controls

Excess fissile material shall not be construed to be "In-Process" to circumvent the requirements of normal storage.

Fissile material shall not be stored in shipping containers for the purpose of negating the requirements of normal storage.

All containers shall be marked or coded to indicate the type, amount, or degree of enrichment.

Containers shall be securely closed and so positioned to prevent significant displacement and maintain criticality prevention requirements.

Container design shall be appropriate for the form of stored material. Criteria for container integrity shall be developed in the course of the required safety analysis and the application of these criteria ascertained by periodic inspection.

Inventory

Up-to-date inventories shall be maintained for the FMCA as a whole, and all individual storage or processing positions except where physical restrictions are such that it is not possible to exceed the applicable limits (e.g., where only one fuel unit will fit into a slot). Appropriate totals shall be maintained to aid in ensuring mass limits compliance.

Nonfueled Assemblies and Experiments

Special precautions are necessary to prevent criticality limit violations when nonfueled components (hardware and assemblies) are similar or identical to fueled components. To preclude handling fissile material inadvertently, nonfueled components shall be treated as if they contain fissile material, unless clearly identified (e.g., tagged or color coded) as being nonfueled. Examples of such nonfueled components are dummy fuel plates and assemblies, nonfueled experiments, and experiment hardware with the experiments removed.

Nonfissile Material

Moderating and reflecting materials, e.g., beryllium, and equipment not requiring storage in the FMCA or not shown to be safe by the Safety Analysis shall be excluded from the FMCA.

Inspections

Equipment essential to criticality safety, as determined in the Safety Analysis, shall be controlled in accordance with the following, as applicable:

- Design of new equipment, essential to criticality safety, shall be submitted to formal design review, as specified by the Quality Manual.

- New equipment shall be inspected, prior to initial use, to ensure that it meets the inspection requirements in the Criticality Safety Evaluation. Existing equipment shall be inspected at periodic intervals if determined desirable by the Safety Analysis. The inspection requirements shall be included in the FMCA control procedures.

- Damaged or defective equipment shall be removed from service, replaced or repaired. Prior to further use, the equipment shall be reinspected.

- Equipment drawings and inspection records shall be maintained in accordance with facility Standard Practices.

Emergency Conditions

Criticality safety control procedures shall include responsibilities and methods for safe operation under emergency conditions.

Criticality Alarm Systems

A criticality alarm system is required at FMCAs wherein the quantities of fissile materials may exceed 700 g uranium-235, 520 g uranium-233, 450 g plutonium, or 450 g of any combinations of these three isotopes. This requirement may be waived, if approved by the Health and Safety Division, for an FMCA having adequate shielding to protect personnel. The criticality alarm system shall include remote alarm annunciators at a location attended 24 hours every day.

FMCA Closures

When an FMCA will no longer be used to store or handle fuel, it shall be officially closed by:

- Removing the fissile material[a]

- Rescinding or withdrawing the control procedures (e.g., Standard Practices)

- Removing FMCA signs

---

a. Less than 15 g could remain in accordance with "Procedures", Page 3.

* • Formally notifying the custodians, Fissile Material Handlers, Safeguards and Materials Management Branch, and Health and Safety Division.

* After the applicable Operational Safety Requirements Document has been rescinded or changes approved, the Health and Safety Division will send a revised list of active FMCAs to FMCA managers, the Safeguards and Material Management Branch, and the Hazardous Materials Shipping Coordinator.

REFERENCES

1. Safety Manual 4030, Safety Qualification Courses.

2. Safety Manual 9010, Administrative Requirements for Fissile Material.

3. Safety Manual 9030, FMCA Safety Analysis.

4. Safety Manual 9040, Criticality Safety Review and Approval.

5. Safety Manual 9070, Shipping and Receiving Fissile Material.

* 6. Safety Manual 2040, Safety Analysis and Review for EG&G Operations.

* 7. DOE Order 5480.1A/ID Order 5480.1, Chapter V, Safety of Nuclear Facilities.

8. DOE Order 5630, Control and Accountability of Nuclear Materials.

APPENDIX A

FISSILE MATERIAL SIGN

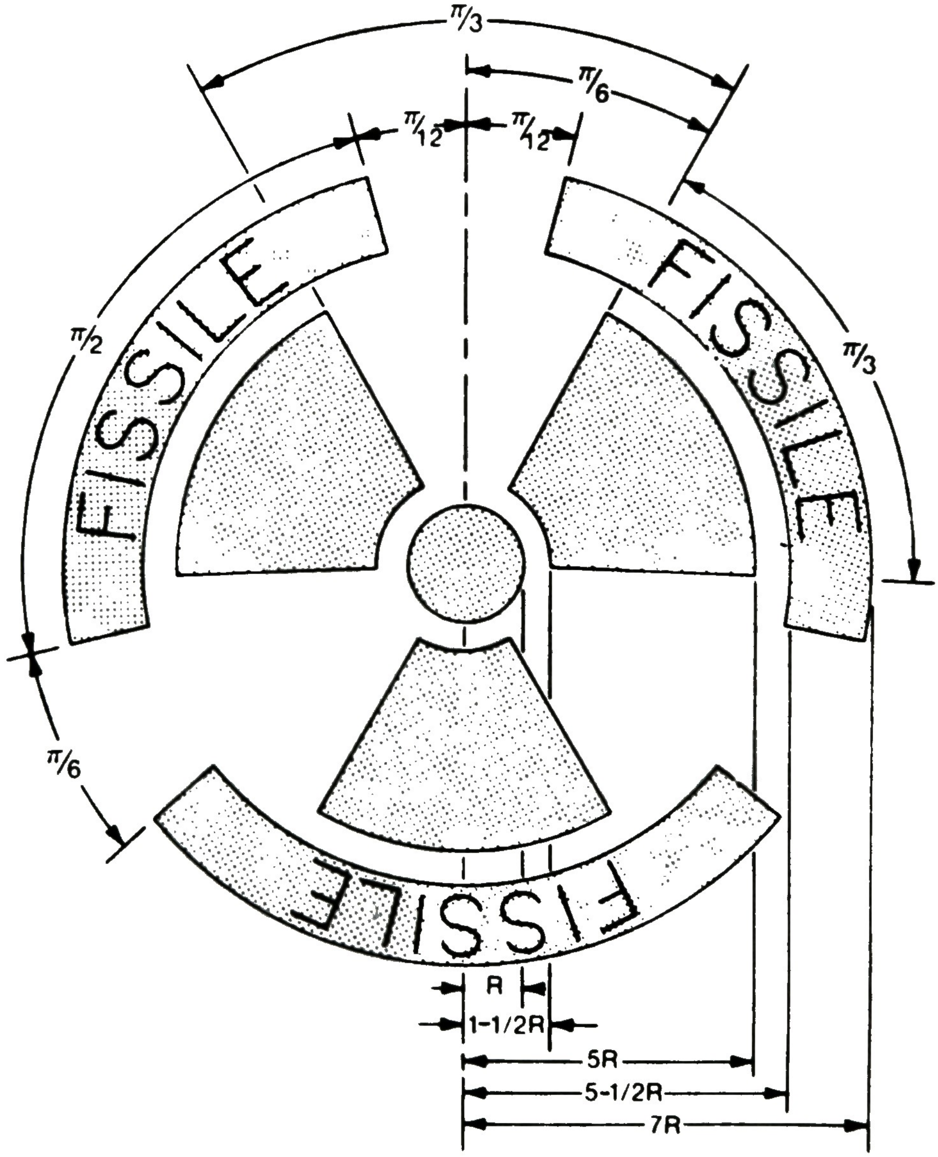
Fissile material signs shall be approximately 0.25 x 0.3 m (10 x 12 in.) in size. This sign shall look like the sign shown on the following page, except that dimensions should be deleted. The letters in the sign shall be block style capital letters of at least 0.025 m (1 in.) in height. The letter width shall be approximately 80% of the letter height, and the stroke width-to-height ratio shall be approximately 1:8.

All shaded areas shall be purple (magenta) and the background shall be yellow. The letters of "FISSILE" in the purple area shall be yellow and equivalent to the "R" dimension in height.

NO WATER SIGN

Where water addition could result in a criticality incident as determined by a safety analysis, a separate sign stating "No Water Area" shall be posted directly below the FMCA sign. The "No Water" sign shall be 0.08 m (3 in.) high by 0.3 m (12 in.) long. The letters shall be 0.04 m (1-1/2 in.) high with the same width and stroke specification as the FMCA sign and shall be purple on a yellow background.

FISSILE MATERIAL



CONTROL AREA

INEL-A-4206

EG&G Idaho Inc

SAFETY MANUAL

Title

FMCA SAFETY ANALYSIS

Approved:

Legend
*Revision
#Addition

(This revision involves updating organization titles and minor editorial changes only)

This document establishes requirements and guidelines for preparing Safety Analyses for FMCAs (fissile material control areas).

SAFETY ANALYSIS

A Safety Analysis (SA) shall be prepared for each FMCA in which more than 200 g of fissile material[a] (400 g if only uranium-235 is involved) is to be handled, processed, or stored prior to the startup of such operations with fissile material. The SA shall be updated, either by a revision or supplement, prior to any FMCA activity change which is outside the existing SA safety envelope. The initiation and funding of the SA is the responsibility of the line manager.

The SA shall consist of two parts:

- A complete evaluation called the CSE (Criticality Safety Evaluation)

- An ICA (Independent Criticality Analysis); an independent calculation or the use of handbook data to verify the calculations in the CSE. This independent calculation is

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a. ^{233}U, ^{235}U, ^{238}Pu, ^{239}Pu, and ^{241}Pu, ^{237}Np, ^{241}Am, and ^{244}Cm.

intended to reveal computer-calculational and usage-of-handbook errors which are difficult to detect except by an independent duplication.

The safety margins used to determine limits must be described in the report. Cumulative safety margins must provide allowance for experimental and computational uncertainties. Possible procedure violations also shall be considered.

* The Health and Safety Division shall prepare CSEs; the Physical and Biological Sciences Division shall prepare ICAs, unless special arangements are made with the Health and Safety Division.

Safety Analyses shall be reviewed and approved, in accordance with SM-9040.

SAFETY ANALYSIS REQUESTS

* Requests for SAs shall be submitted to the Health and Safety Division in writing, either by a Work Release or letter, including a charge number.

Emergency requests for SAs may be made orally by line management, but must be confirmed by a written request.

All requests should include a desired completion date based on need and anticipated complexity of the SA.

The following information shall be provided with the request:

- Title (descriptive of particular evaluation)

- Clear statement of problem or information desired

- Detailed problem description, including:

  - Description of geometry and array of the fissile material and associated equipment (including sketches and applicable drawings)

  - Detailed description of the fissile material (enrichment, density, and if a mixture, isotopic composition, weight percentages)

  - Detailed description of the materials accompanying fissile materials (impurities, cladding, coating, etc.)

  - Detailed description (composition, density, weight percentages, geometry) of the vessels containing the fissile material

  - Description of the environment of the fissile material and associated equipment to a distance of about 3 m in all directions (cells walls, floor, etc.)

  - Facility susceptibilities to earthquake, fires, flooding, etc.

- Description of normal activity or process, and possible deviations therefrom

- Available critical experimental data (if any)

- Any other information which the requestor thinks would aid the Criticality Safety specialist in developing a thorough understanding of the situation.

The Health and Safety Division shall request the ICA from the Physical and Biological Sciences Division. (Line management funds shall be used for the preparation of the ICA.)

REFERENCES

1. Safety Manual 9040, Criticality Safety Review and Approval.

2. Safety Manual 2040; Safety Analysis and Review for EG&G Operations.

FORM 9340-684 (Rev 01-81)

SM- 9040, Rev. 3

EG&G Idaho Inc

SAFETY MANUAL

Title: CRITICALITY SAFETY REVIEW AND APPROVAL

Approved:

Legend
*Revision
#Addition

(This revision involves updating organization titles and minor editorial changes only)

This manual section establishes the responsibilities and requirements for obtaining an independent review of control procedures (Standard Practices) and Safety Analyses relating to processing, handling, shipping, and storing fissile material.

INDEPENDENT REVIEW AND APPROVALS

All new and revised criticality safety control procedures and Safety Analyses shall have independent review. The review process is shown in Appendix A.

CRITICALITY REVIEW COMMITTEE (CRC)

* A CRC (Criticality Review Committee) shall be appointed by the Health and Safety Division Manager to act in an advisory capacity to him. This committee shall consist of a chairperson and two members (who are designated alternate chairpersons) and two alternate members. At least one member shall have a working knowledge of criticality physics, and at least two of the members shall be non-Health and Safety Division employees.

* The CRC shall review all safety analyses. Materials submitted shall have evidence of approval of the cognizant line management and of initial review by Criticality Safety, Health and Safety Division.

The CRC shall consider the following in making their review:

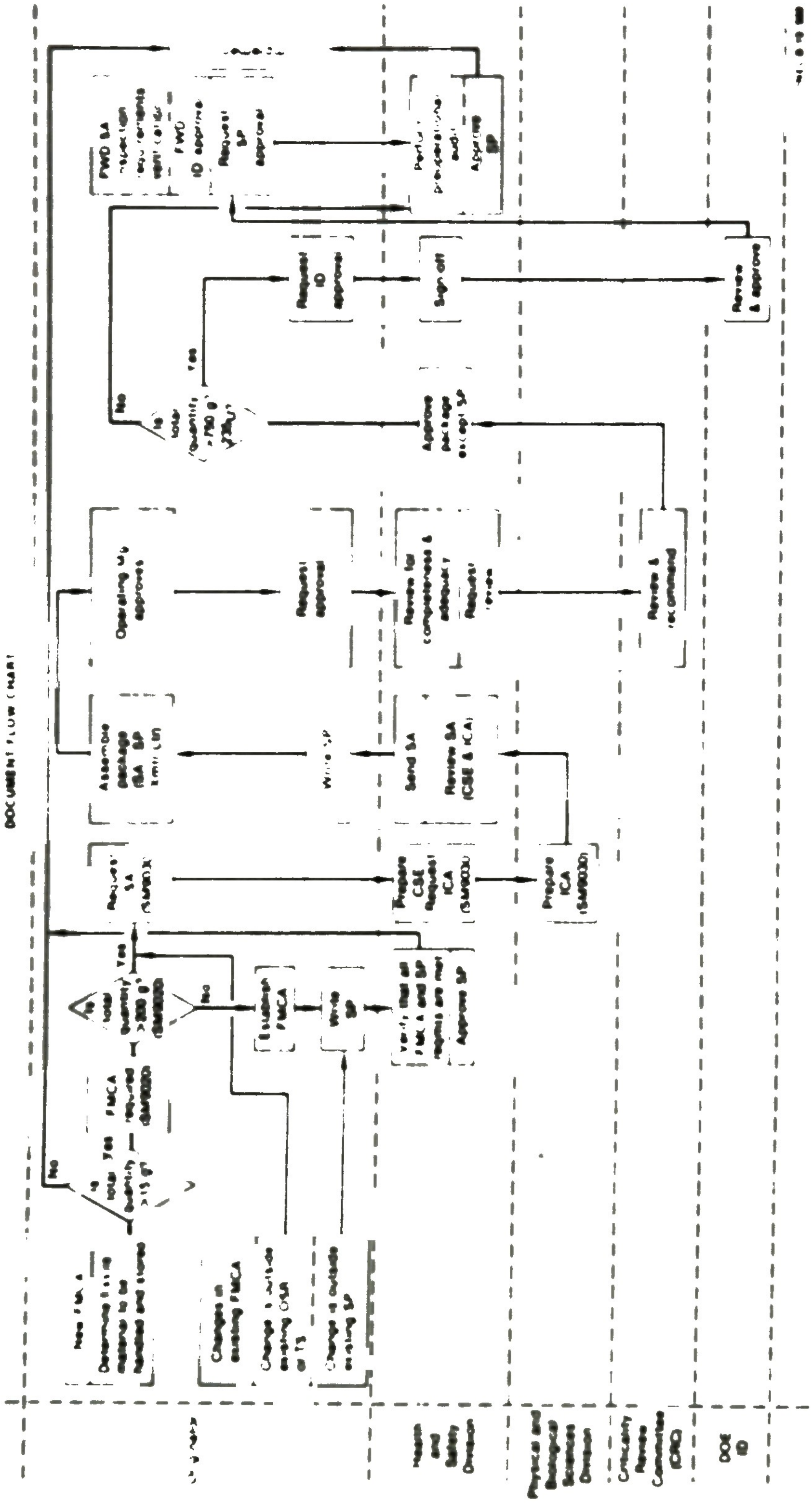
- Technical adequacy
- Validity of approach
- Calculation adequately verified
- Administrative and physical controls adequate
- Implementation feasible.

REFERENCES

* 1. DOE Order 5480.1A/ID Order 5480.1, Chapter V, Safety of Nuclear Facilities.

2. Safety Manual 9030, FMCA Safety Analyses.

APPENDIX A



DOCUMENT FLOW CHART

EG&G Idaho, Inc

SAFETY MANUAL

Title: SHIPPING AND RECEIVING FISSILE MATERIALS

Approved: [signature]

Legend
*Revision
#Addition

(This revision involves updating organization titles and minor editorial changes only)

This document establishes criticality control requirements for shipping and receiving fissile material.

* SAFETY RESPONSIBILITIES

* LINE MANAGERS responsible for shipping or receiving quantities of fissile material[a] greater than or equal to 15 g shall:

Maintain configuration control over the fissile material in accordance with SM-5070, SM-5080, and the following instructions:

* • Materials packaged for transport, which comply with the DOT classification (Fissile Class I, II, or III), may be temporarily stored in the DOT-approved shipping containers outside FMCAs, provided the Transport Index is less than 50 and the storage meets the security requirements specified by the EG&G Safeguards and Materials Management Branch.

* • Materials stored temporarily outside FMCAs, under the requirement above, shall be separated by at least 4 m (12 ft) from all other fissile material. This separation shall be maintained throughout the period of temporary storage. The shipping containers shall be clearly marked (legible for a distance of 4 m) as containing fissile material.[a] The Health and Safety Division shall be notified of the intent of such

a. ^{233}U, ^{235}U, ^{238}Pu, ^{239}Pu, and ^{241}Pu, ^{237}Np, ^{241}Am, and ^{244}Cm.

temporary storage and shall give at least oral consent before storing the material. No more than one approved shipping container may be taken into a FMCA for temporary storage or unloading, unless such handling is performed in accordance with control procedures specifically approved for that operation in that FMCA.

- Receivers of fissile material shall verify the contents of each fissile material shipment received prior to storing the material in a FMCA. Verification must be documented and will be based upon a piece count, check of serial numbers, material descriptions, or actual physical verification by chemical or spectral analysis. Discrepancies found during this verificaton process shall be reported to the EG&G Safeguards and Materials Management Branch.

- Abnormal or damaged shipments, or shipments which cannot be stored within current FMCA procedures, shall be temporarily stored in accordance with the first and second requirements above, until a disposition procedure is prepared, reviewed, and approved.

- Fissile material transfers between FMCAs (within or between INEL facilities) require the mutual concurrence of the cognizant custodians or supervisors. Transfers of quantities in excess of 200 g of fissile material[a] (400 g if only uranium-235 is involved) must be made under one of the following conditions:

---

a. ^{233}U, ^{235}U, ^{238}Pu, ^{239}Pu, and ^{241}Pu, ^{237}Np, ^{241}Am, and ^{244}Cm.

- In approved shipping containers, in compliance with SM-5070

* - In compliance with a Health and Safety Division-approved control procedure specially prepared for the transfers in question

Quantities of fissile material less than those permitted outside of storage, as specified in the approved FMCA control procedures for both the sending and receiving areas, may be transferred within a plant area without specific approval, providing the FMCA control procedures for both areas are not violated and the fissile material is constantly attended by a person who has been trained in nuclear criticality safety, as required in SM-4030.

REFERENCES

1. Safety Manual 5070, On-Site Radioactive Shipments.

2. Safety Manual 5080, Off-Site Radioactive Shipments.

3. Safety Manual 5090, Reusable Packages for Radioactive Material Shipments.

4. Safety Manual 4030, Safety Qualification Courses.

APPENDIX B

CRITICALITY CONSIDERATIONS

EG&G Idaho, Inc.

FORM EG&G-2631 (Rev. 4-78)

ENGINEERING DESIGN FILE

PROJECT FILE NO

EDF SERIAL NO TMI-5

FUNCTIONAL FILE NO ADS-6 -82

DATE November 22, 1982

PROJECT TASK

SUBTASK Criticality Considerations

EDF PAGE NO ___ OF ___

SUBJECT TMI Fuel Canister Criticality Safety Design Criteria

ABSTRACT

Refs: (a) American National Standard, Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors, ANSI N16.1- 1975.

(b) DOE Order 5480.1R, Chapter V, Safety of Nuclear Facilities, August 13, 1981.

(c) DOE Order 5480.1 Supplement, Chapter V, Safety of Nuclear Facilities, July 27, 1982 (Idaho Division).

Criticality safety provisions shall meet the requirements of References (a) through (c). More specifically:

1. Essential criticality safety design features must be verified and documented. This includes canister size, canister material, and any fixed neutron poison.

2. Considerations of 'future use' of handling, transporting, and storage of canisters shall be included in the canister design. This includes neutron interaction between canisters located in an array. Also the ability to periodically inspect any fixed neutron poison.

3. Computer codes and cross section sets used for calculations must be validated against applicable critical experiments.

4. Off normal conditions must be considered in accordance with the 'double contingency principle'. Some of these conditions are:

   - Overbatching of fissile material
   - Loss of solid neutron absorber
   - Redistribution of neutron absorber and fissile material
   - Change in canister dimensions or breach of containment from lack of structural integrity. Possible mechanisms could be corrosion or gas pressurization.
   - Loss of array neutron isolating material such as storage basin drainage accident.

(continued)

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ADS A. D. Summers	SH&S	DW Knight	11-23-82	BL Rich	11/23/82

- Credible moderation. This might include any moderator materials (Be, C, etc.) that may be mixed with core debris or influx of storage basin water.

5. Reflectors more efficient than water (lead cask wall, concrete storage basin wall and floor, etc.) shall be considered.

6. In the criticality calculations, the highest enrichment instead of an average enrichment, shall be used.

Note: Any later ANSI standards or DOE orders will supplement the referenced documents and will apply.

APPENDIX C

CUMULATIVE RADIATION EXPOSURE

EG&G Idaho, Inc

FORM EG&G-2631 (Rev. 4-78)

ENGINEERING DESIGN FILE

PROJECT FILE NO

EDF SERIAL NO RA-84-20

FUNCTIONAL FILE NO

PROJECT TASK

DATE May 11, 1984

SUBTASK

EDF PAGE NO ___ OF ___

SUBJECT CUMULATIVE RADIATION EXPOSURE OF TMI-2 FUEL CANISTER COMPONENTS DURING STORAGE IN THE TAN-607 WATER PIT

ABSTRACT

Refs: (a) P. G. Voilleque, Estimated Source Terms for Radionuclides and Suspended Particles During TMI-2 Defueling Operations, GEND-INF-019 (1982).

(b) TMI-2 Accident Core Heat-up Analysis NSAC-25 (1981)

Certain components of the TMI-2 fuel canisters could be sensitive to radiation damage during long term storage in the TAN-607 water pit, depending on the choice of materials for the gaskets and/or O-rings used with the removable closure heads or in the venting mechanisms. The magnitude of the radiation fields to which these devices will be subjected will dictate the choice of materials and a potential replacement schedule.

Although a final decision has not been made on the canister dimensions and the loading parameters, it is still possible to provide estimates of the cumulative radiation exposures likely to be encountered by the gasket seals. Obviously these radiation fields will be very dependent on both distance from the fuel and on shielding and these, in turn, will be dependent on the canister dimensions and design.

Two estimates of the radiation fields have been calculated based on two different assumed canisters and loading parameters. Loading pattern A assumes the total TMI-2 core will be pulverized and approximately 800 kg of core debris will be loaded into each canister. For loading pattern A, the canisters are assumed to be cylinders 10 inches in diameter and 10 feet in length. The 800 kg of debris would then occupy the lower 6 feet leaving 4 feet of void for shipping. The canister would be flooded with water for storage in the TAN-607 water pit. The closure head is assumed to provide at least 1/4 inch of steel shielding for the gaskets in addition to 4 feet of water.

Loading pattern B assumes cylindrical canisters 15 feet in length with cross-sectional areas capable of accepting an intact TMI-2 fuel assembly (710 kg and 13.8 ft. in length). Again the closure head provides at least 1/4 in. steel shielding in addition to 6 in. of water shielding.

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AUTHOR	DEPT	REVIEWED	DATE	APPROVED	DATE
D. E. Martz	5	P.E. Ruhter	5/11/84		

Table 1. Partial Listing of Calculated Radionuclide Activities for TMI-2 Shutdown and after Decay

Radionuclide	Half-Life[b] (yr)	Radionuclide Activity (Ci)[a] At Shutdown (t_d = 0)	After Decay (t_d = 63 months) July 1984	After Decay[d] (t_d = 93 months) Jan. 1987
^{3}H[c]	12.33	4.1×10^3	3.1×10^3	2.7×10^3
^{85}Kr	10.7	9.7×10^4	6.9×10^4	1.4×10^4
^{90}Sr-Y	28.8	7.5×10^5	6.6×10^5	6.2×10^5
^{106}Ru-Rh	1.01	3.3×10^6	9.0×10^4	1.7×10^4
^{125}Sb	2.7	1.2×10^5	3.3×10^4	8.9×10^3
^{134}Cs	2.062	1.6×10^5	2.7×10^4	1.2×10^4
^{137}Cs	30.17	8.4×10^5	7.5×10^5	7.1×10^5
^{144}Ce-Pr	0.778	2.5×10^7	2.3×10^5	2.5×10^4
^{147}Pm	2.6234	2.6×10^6	8.1×10^5	4.2×10^5
^{151}Sm	90	1.1×10^4	1.1×10^4	1.0×10^4
^{155}Eu	4.9	3.2×10^4	1.5×10^4	1.1×10^4
^{238}U	4.468×10^9	2.7×10^1	2.7×10^1	2.7×10^1
^{238}Pu	87.74	7.3×10^2	7.6×10^2	7.5×10^2
^{239}Pu	2.41×10^4	8.6×10^3	9.0×10^3	9.0×10^3
^{240}Pu	6.57×10^3	2.4×10^3	2.4×10^3	2.4×10^3
^{241}Pu	14.4	2.0×10^5	1.6×10^5	1.4×10^5
^{241}Am	433	2.1×10^1	1.9×10^3	1.9×10^3
		3.3×10^7 Ci	2.87×10^6 Ci	

a. The quantity of t_d is the decay time.

b. Half-lives were taken from and are given with the same number of significant figures as in Reference 3.

c. An additional 200 Ci is estimated to have been produced by neutron activation reactions in the coolant during power operation.

d. Calculated by D. E. Martz

(From P. G. Voilleque, GEND-INF-019, 1982)

It should be noted that self absorption of both betas and gammas in dense fuel is very effective and consequently the radiation doses at the surfaces are nearly the same for a small quantity of fuel as for a much larger quantity. Because of the voids and open passages in an intact fuel assembly, the gamma dose may actually be larger at the end surface than for a compact mass of fuel. In these estimates the dose rate was calculated for a compact cylindrical 800 kg mass of TMI-2 fuel debris using the ISOSHLD-II program and the Voilleque source terms in Table 1 as of July 1984. Hand calculations were made to determine the relative contributions from each isotope and the effective half lives of the gamma and beta activities separately.

Table 2 lists the interpolated gamma dose rates for different distances from the fuel source. Since the water and steel shielding is more than sufficient to remove the beta and low energy brehmsstrahlung photons (<80 kev) the dose rates at the gasket locations represent the hard gamma component only. Cs-137 was found to be contributing 86% of the total gamma dose as of July 1984, and the 30.17 year half-life of this isotope controls the gamma activity decay beyond 1984.

TABLE 2. GAMMA EXPOSURE RATES AT VARIOUS DISTANCES FROM 800 kg OF TMI-2 FUEL

Distance and Shielding	Estimated Dose Rate (1984) R/h
1 cm (no shielding)	12,800 (includes brehmsstrahlung)
10 cm (no shielding)	5,400 "
30 cm (no shielding)	1,530 "
90 cm (no shielding)	190 "
Location A (4 ft water + 1/4" steel)	0.002
Location B- (6 in.water + 1/4 " steel)	713
Inside fuel (no shielding)	20,000 R/h beta + 12,800 R/h gamma

At reactor shutdown in 1979, the Ce-144/Pr-144 and Ru-106/Rh-106 fission isotopes were contributing 96% of a calculated 400,000 R/hr beta surface dose. By 1984 these shorter half-life istopes have decayed until they contribute only 26% of a total 20,000 R/h beta surface dose. Sr-90/Y-90 has now become the major beta contributor. From 1984 on the effective beta decay will closely follow the 28.8 y half-life of Sr-90.

Table 3 lists the cumulative hard gamma exposures at the gasket locations for loading patterns A and B at various times after 1984. The extra shielding provided by 4 feet of water greatly reduces the radiation exposure of gasket location A compared to location B with only 6 inches of water. Distance and shielding are so significant that these values must be considered only tentative estimates and the actual values must await the actual canister dimensions and loading pattern.

TABLE 3. CUMULATIVE GAMMA EXPOSURE IN RADS

Time (yrs)	Activity Fraction Remaining	Cumulative Dose (Rads) Location A	Location B
0	1.000	0	0
1	.977	17	1.0×10^7
2	.955	45	2.0×10^7
3	.933	72	3.0×10^7
4	.912	99	3.9×10^7
5	.892	125	4.9×10^7
10	.795	247	9.3×10^7
15	.709	356	13.2×10^7
20	.632	453	16.7×10^7
30	.502	616	22.6×10^7

APPENDIX D

TMI-2 FUEL CANISTER STRONGBACK: PRECONCEPTUAL DESIGN

EG&G Idaho, Inc.

FORM EG&G-2631 (Rev. 4-79)

ENGINEERING DESIGN FILE

PROJECT FILE NO TMI-2

EDF SERIAL NO

FUNCTIONAL FILE NO

PROJECT/TASK TMI Fuel Canister Strongback

DATE November 12, 1982

SUBTASK Preconceptual Design

EDF PAGE NO 1 OF 41

SUBJECT

TMI FUEL CANISTER STRONGBACK

ABSTRACT

This EDF documents a pre-conceptual design study for a TMI fuel canister strongback and upending device. The study shows that a strongback-rotation device can be designed to perform the laydown and transport function for a loaded TMI fuel canister.

Few constraints are imposed on the canister design, however, a list of canister criteria and "desirable features" was generated.

DISTRIBUTION (COMPLETE PACKAGE) R. L. Drexler, D. E. Wilkins, J. M. Bower

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ENGINEERING DESIGN FILE

PROJECT FILE NO. TMI-2
EDF SERIAL NO --
FUNCTIONAL FILE NO

PROJECT/TASK TMI Fuel Canister Strongback
DATE November 12, 1982
SUBTASK Preconceptual Design
EDF PAGE NO OF

1.0 ASSUMPTIONS

1.1 It is assumed that the following equipment may be required for handling TMI core debris canisters in the TAN hot shop (in additon to canister storage racks).

A. Canister Lift Fixture - Remote installation and removal for vertical lift of canisters from cask into strongback or temporary support stand.

B. Underwater Crane Hook Extension - Fixture to keep the hot shop crane hooks, blocks and cables out of the pool water for transfer of racks to and from the storage pool.

C. Strongback and Rotation Device - Receives canister in the vertical attitude, rotates it to a horizontal position, balances assembly for a level attitude, and adapts to the existing hot shop to RML transporter. Fixtures must be suitable for the reverse procedure to return canister to vertical storage attitude.

D. Canister Closure Removal and Installation Tooling - Removes canister closure and replaces the closure for a sealed-for-storage condition. Equipment must function with canister in horizontal attitude in the RML. May include seal cutting and welding functions.

E. Debris Removal and Replacement Tooling - Equipment for removal of core debris material from a canister, and replacement of debris into the canister after sampling and analysis. Equipment must function in the RML, with the canister in a horizontal attitude.

F. Leak Test Equipment - If the re-closed canister must be certified "leak-tight" for long term storage, a remote seal integrity test device may be necessary.

G. Canister Inverting Fixture - If all the material in a canister has to be removed, a fixture to rotate canisters for dumping the contents may be required. (This need may be further in the future.)

1.2 It is assumed that the canisters will be circular in cross section and will be constrained by the shipping cask to 13.38 in. diameter and 15 ft in length.

1.3 It is assumed that the canister lifting fitting at its upper end should not be used for the laydown/upending operation.

1.4 It is assumed that the canister may need lateral support during laydown and upending operations.

2.0 STRONGBACK ASSEMBLY CONCEPT

A pre-conceptual design of item 1.1.C, Strongback and Rotation Device, has been formulated for estimation purposes. This fixture, consisting of a base assembly and a strongback assembly could be designed and built for use in the TAN 607 hot shop. The fixture would be compatible with the hot shop crane, manipulators and hot cell transporter system, and would use either the 10 ton or 100 ton hot shop crane hoist for the laydown and upending operations. Either the O-man or wall mount manipulators could be used for the fixture horizontal balance adjustment. Either the 10 ton or 100 ton crane hoists could be used for lifting the assembly in the horizontal attitude.

The strongback has a support base which engages the transporter deck and seismic restraint hooks. It also has feet which allow it to rest on the hot shop or RML floor without the base assembly. If necessary, it could be made of stainless steel and designed for underwater pool storage, however, the present concept is for carbon steel construction and dry operation only.

General arrangement of the Laydown Fixture Assembly is shown in Figure 1. The fixture barrel is sized to receive the fuel canister and support it during the laydown sequence, during the transport from hot shop to RML cell, and while in the RML cell.

3.0 BASE ASSEMBLY CONCEPT

3.1 The base assembly has no moving parts and would be pre-positioned in the shop in a location advantageous for viewing and manipulator access. The base assembly which requires no attachment to the facility, also serves as a storage stand for the laydown fixture when it is not in use. A latch could secure the base and strongback together for handling and moving the two assemblies as a unit.

3.2 The base assembly concept is shown in Figure 2.

4.0 CANISTER LAYDOWN OPERATING SEQUENCE

4.1 A possible sequence of the laydown operation is shown in Steps A through M.

4.2 Initial Conditions - Fixture strongback standing on end supported by base assembly, and canister suspended from crane.

4.3 Operating Sequence:

A. Lower the canister into the upright strongback tube.

B. Disengage crane from canister.

C. Engage crane hook into strongback pivoting bail.

D. Lift strongback from base assembly.

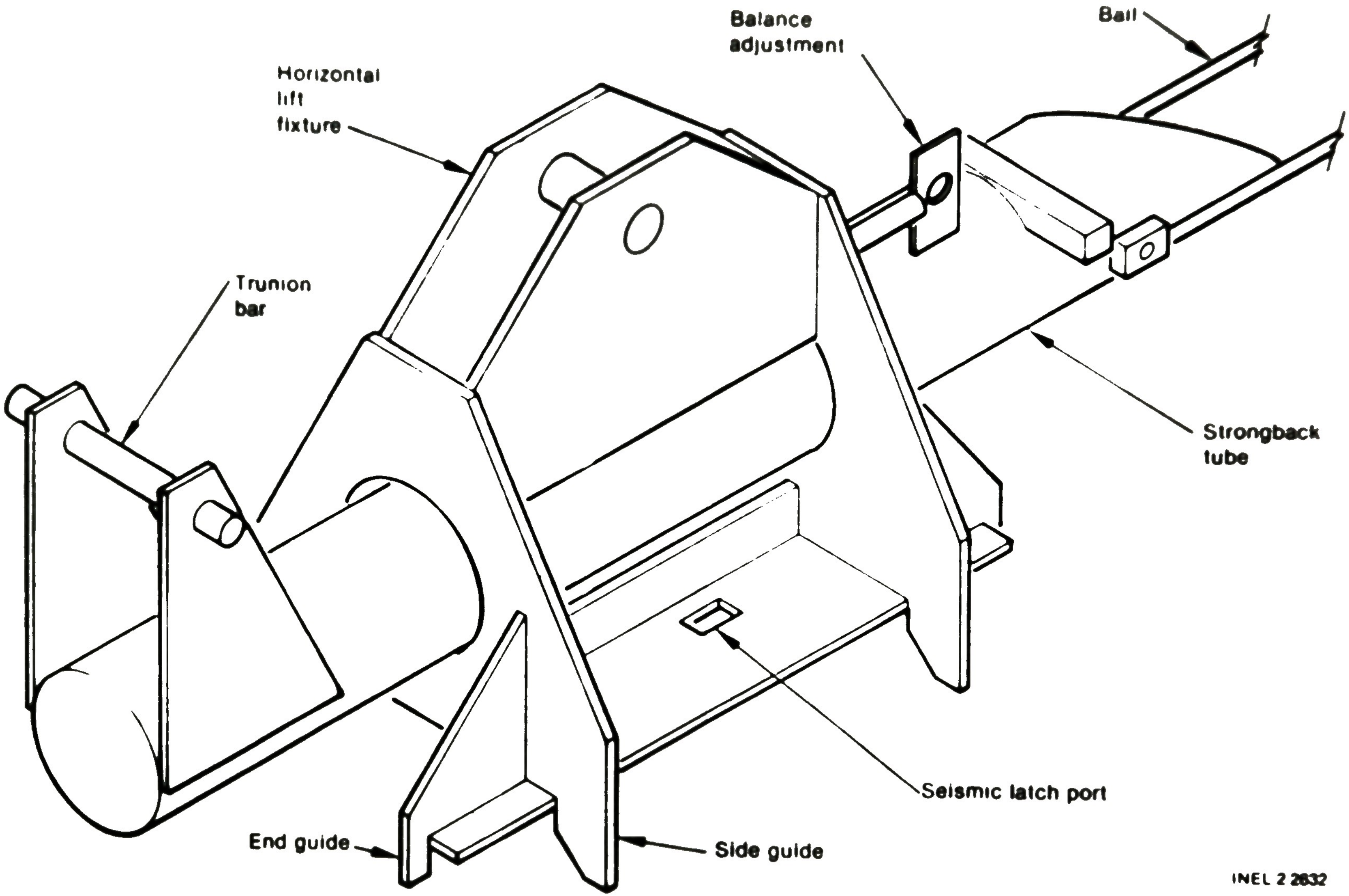


Figure 1. Strongback Assembly

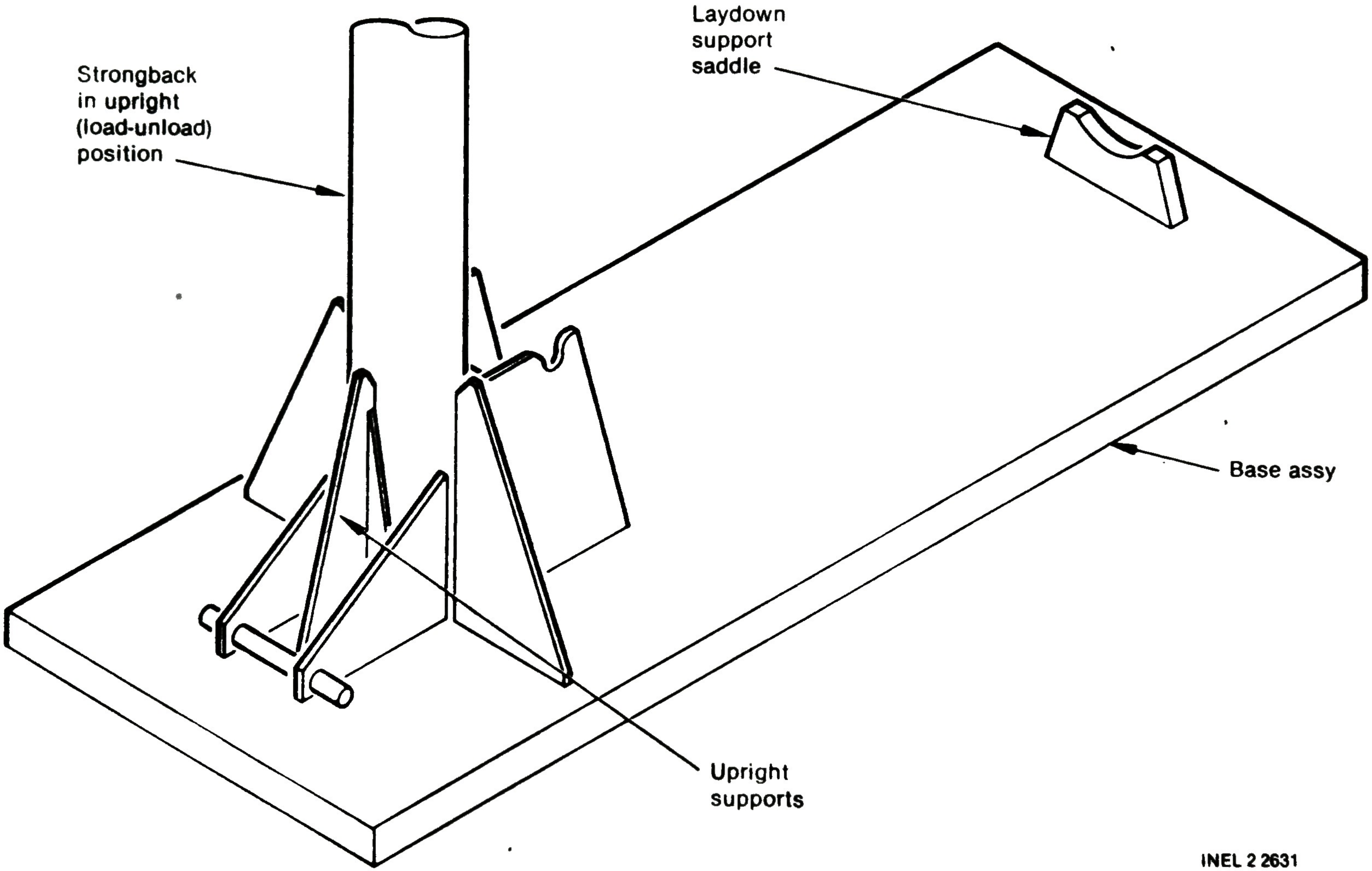
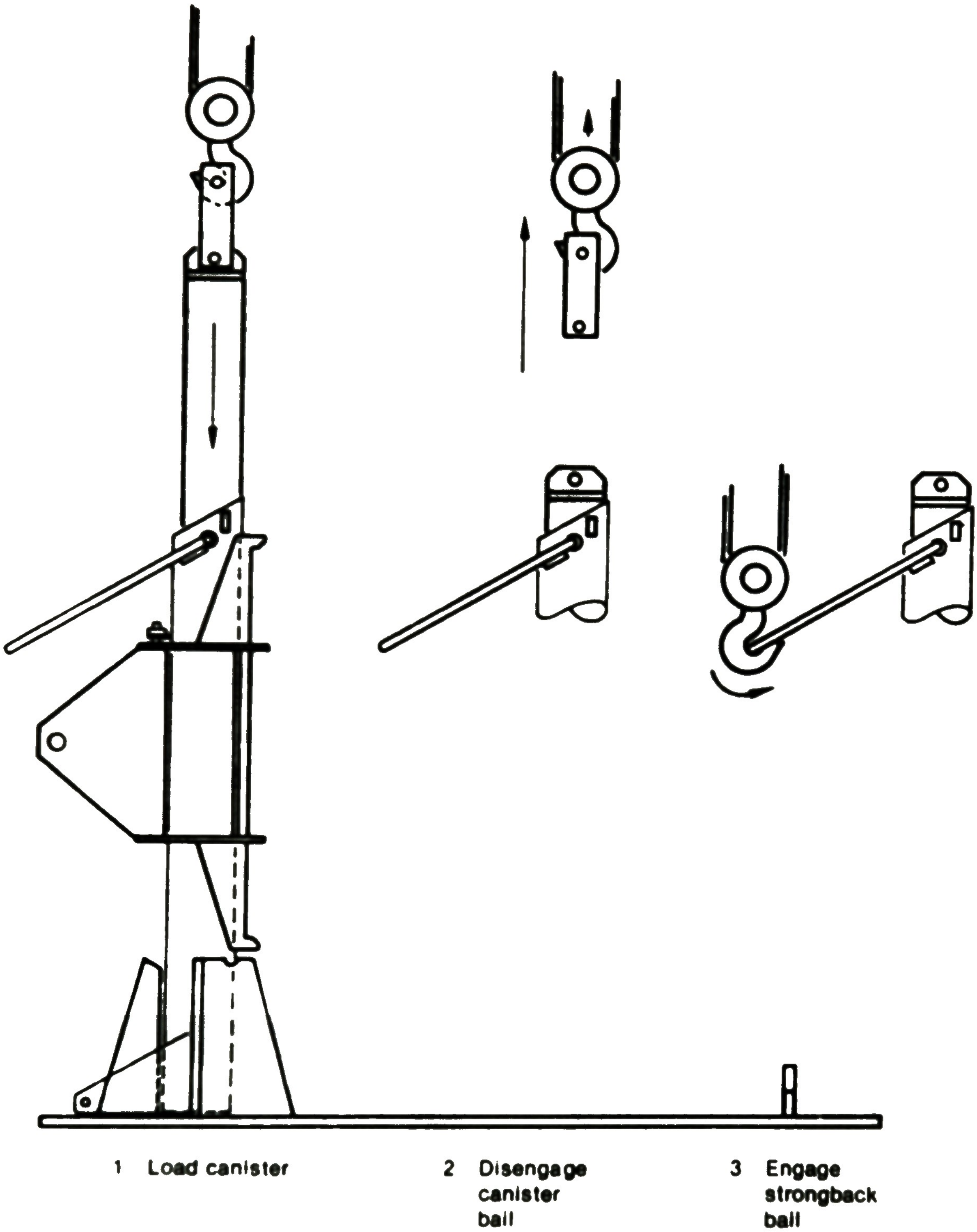
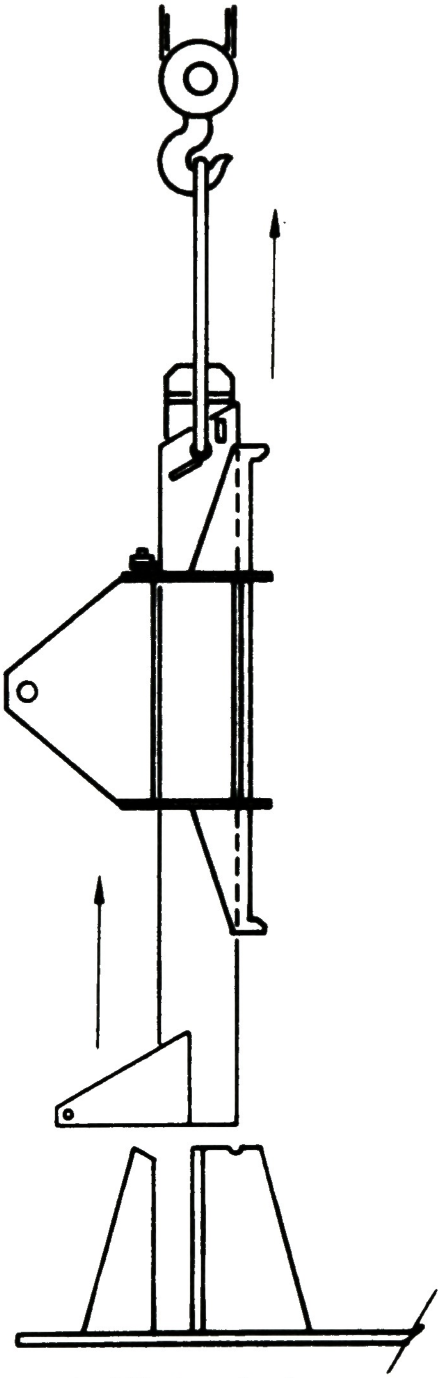


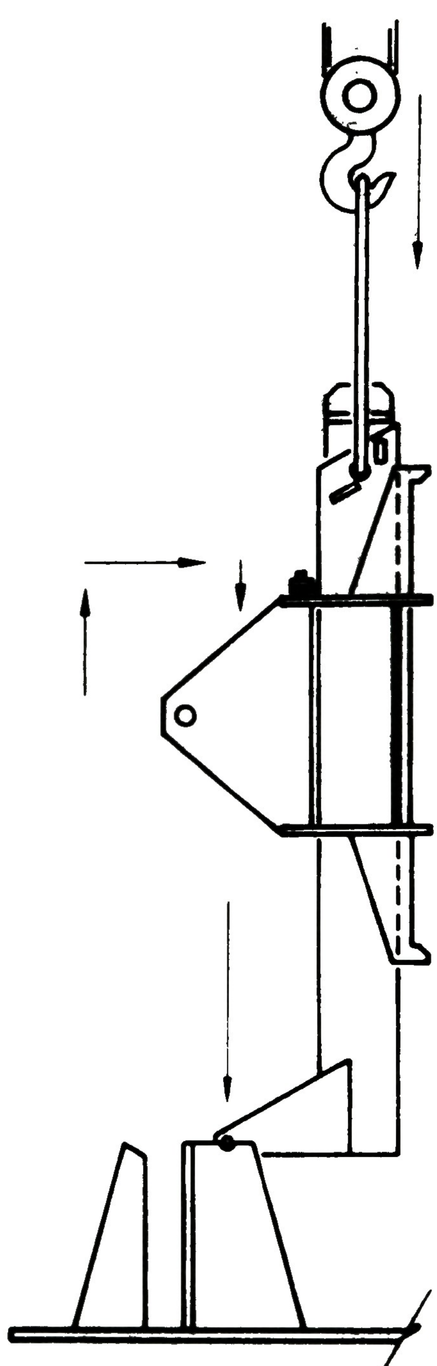
Figure 2. Base Assembly


1 Load canister
2 Disengage canister bail
3 Engage strongback bail

INEL 2 2623

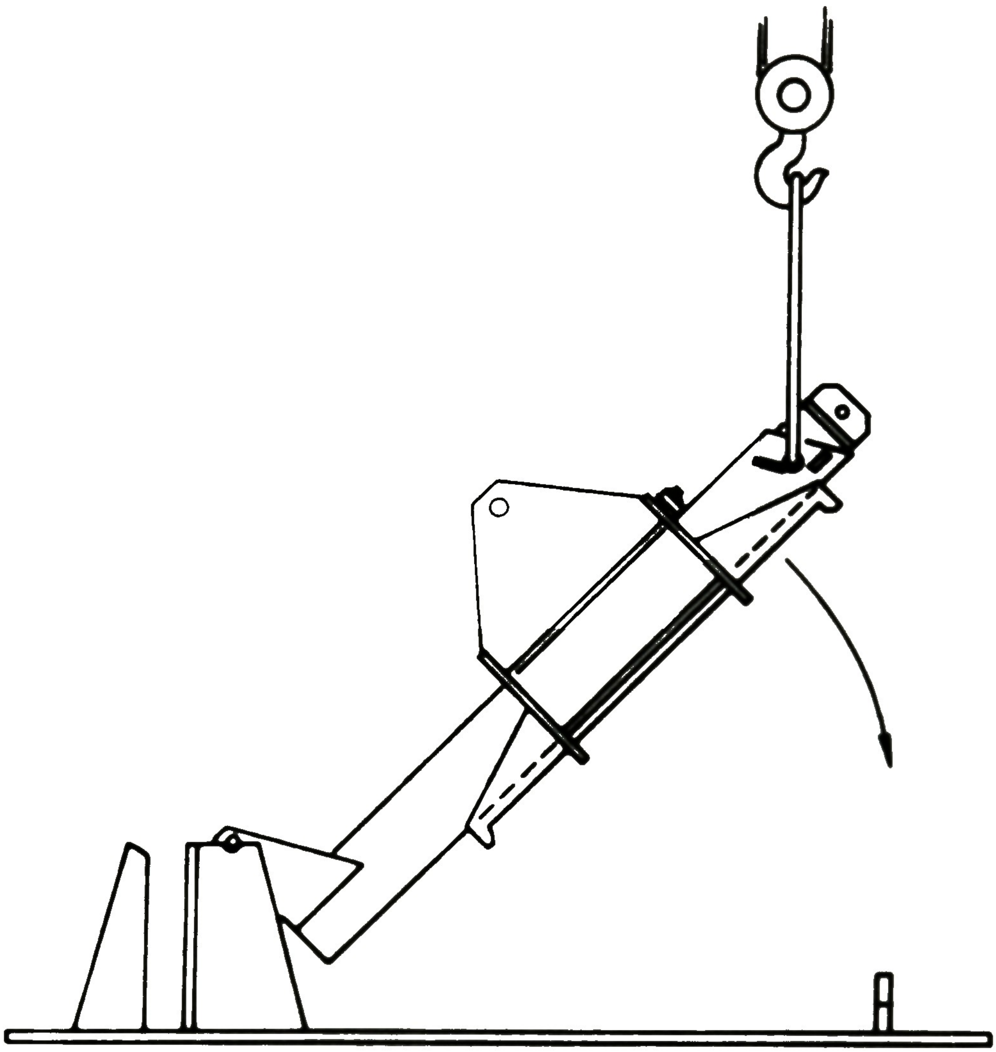


4 Lift strongback from base



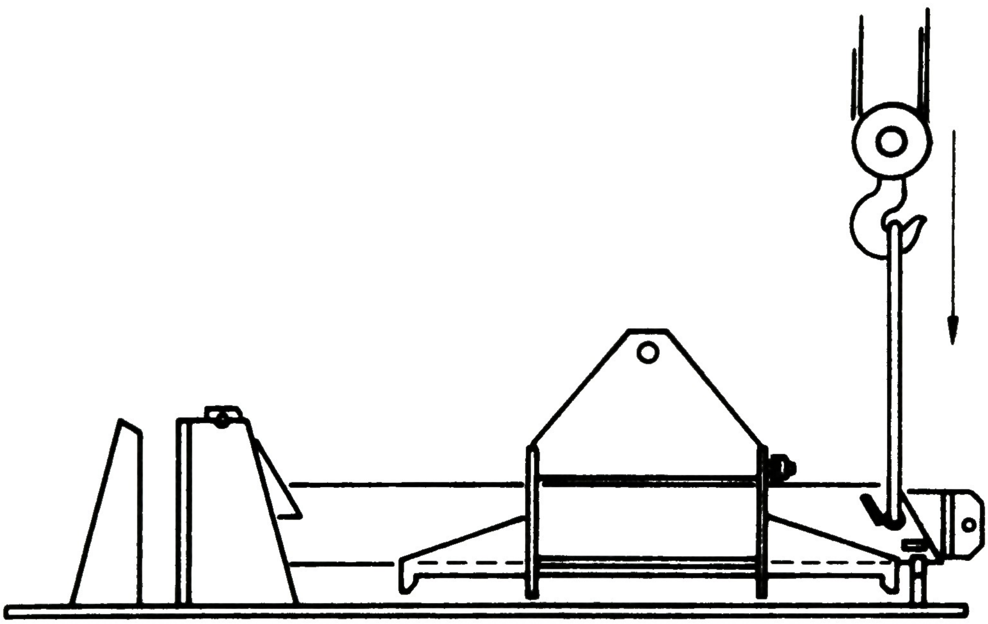
5 Engage trunions

INEL 2 2629

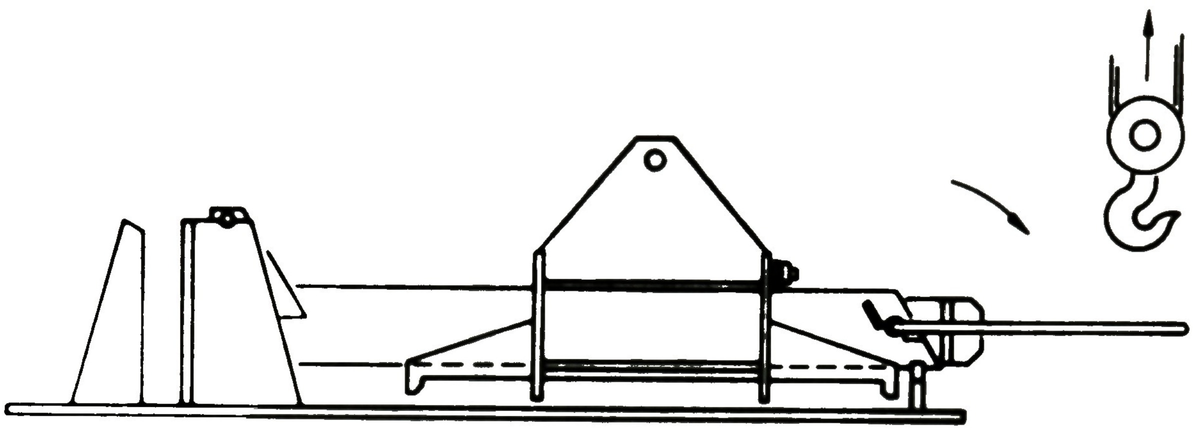


6 Laydown

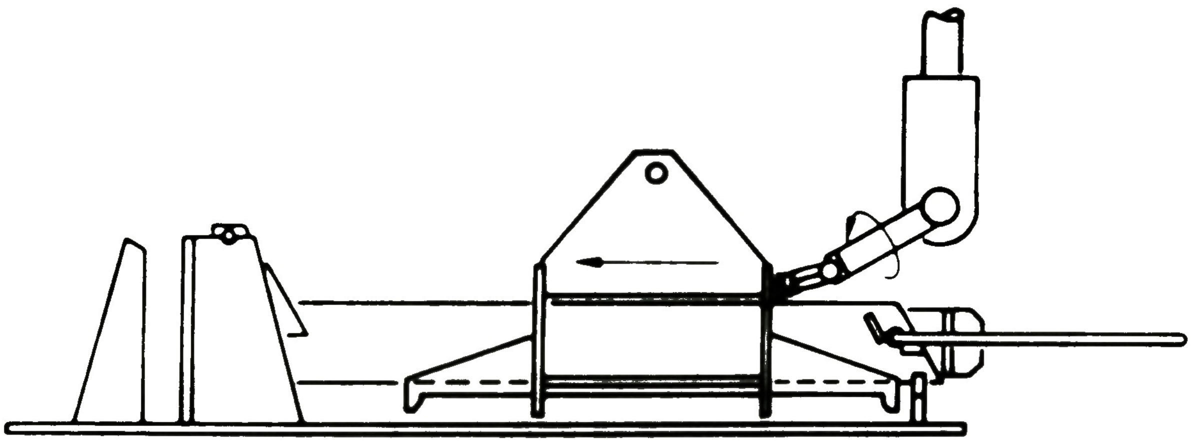
INEL 2 2830



7 Laydown complete INEL 2 2627

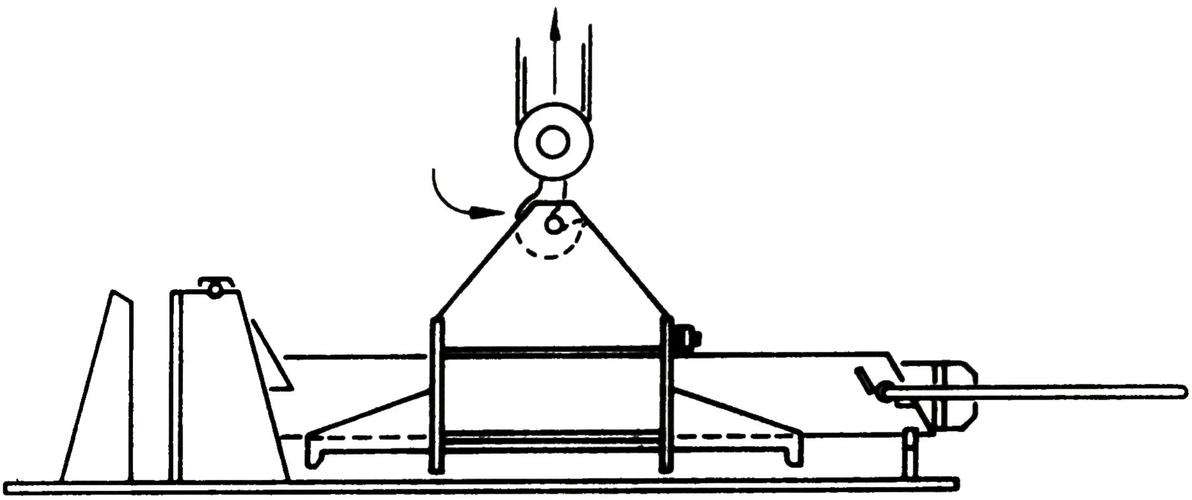


8 Disengage strongback bail

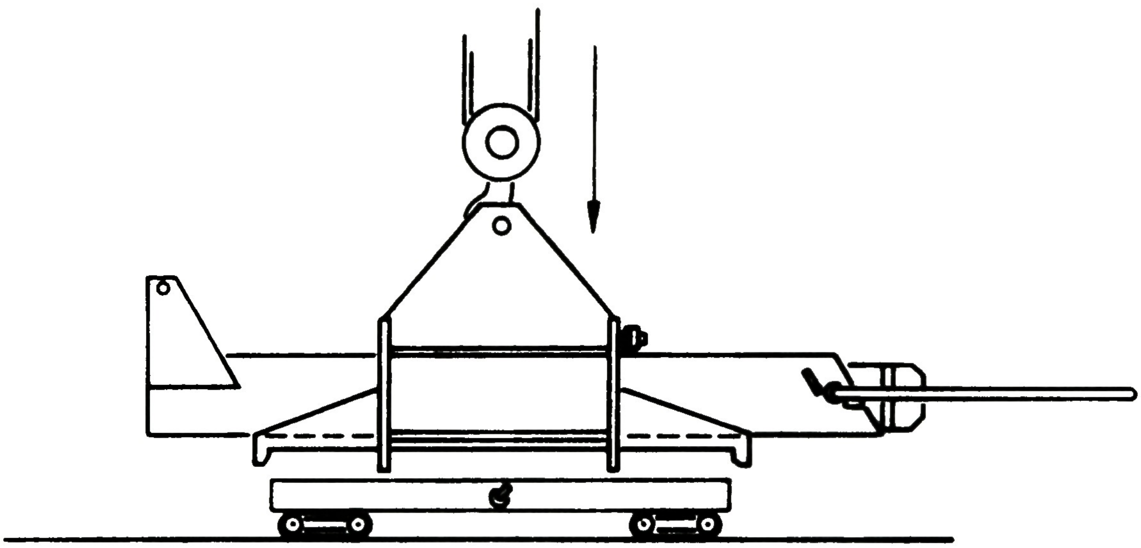


9 Adjust lift fixture balance

INEL 2 2628

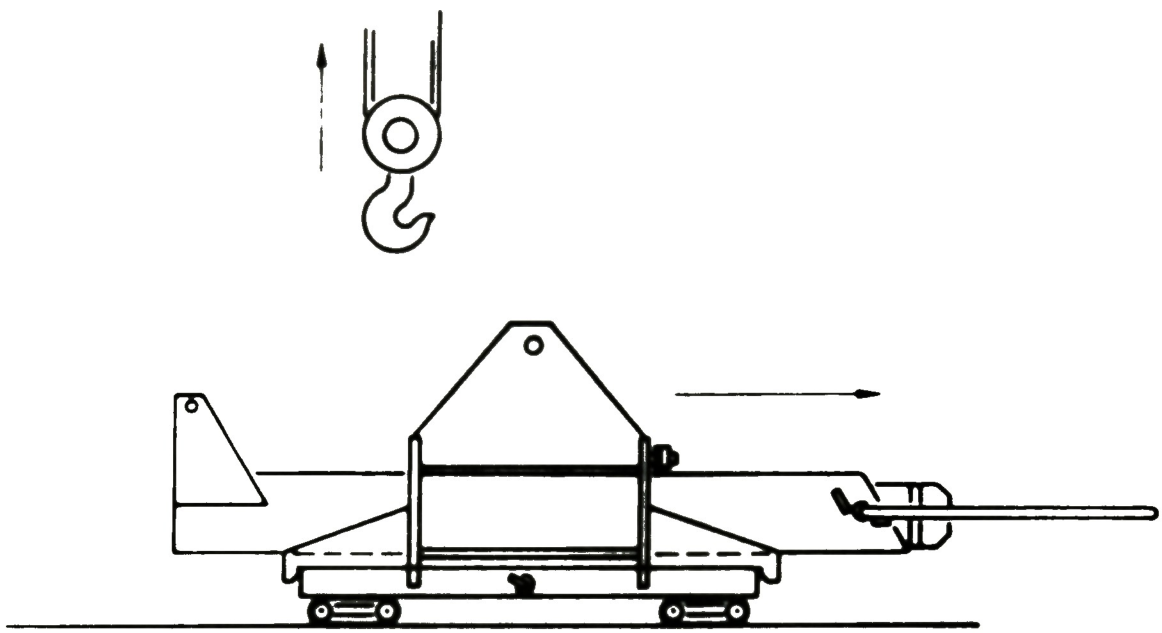


10 Engage lift fixture



11 Lower strongback-assy onto transporter

INEL 2 2626



12 Disengage crane
transport to RML

INEL 2 2625

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FORM EG&G-2631A (Rev. 4-78)

PROJECT FILE NO TMI-2

ENGINEERING DESIGN FILE

EDF SERIAL NO

FUNCTIONAL FILE NO

PROJECT/TASK TMI Fuel Canister Strongback

DATE November 12, 1982

SUBTASK Preconceptual Design

EDF PAGE NO OF

E. Translate and lower strongback to engage pivot trunnions on base.

F. Rotate strongback by lowering and translating crane.

G. Lower strongback onto horizontal support saddle of base assembly.

H. Lower bail to lock canister in strongback, and disengage crane hook from bail.

I. Adjust beam assembly position to estimated center of gravity for balanced lift in horizontal attitude.

J. Engage crane hook into horizontal lift beam and make trial lift for balance. Repeat Step I as required to achieve balanced lift with strongback in horizontal attitude.

K. Move assembly into position over transporter air pallet and lower to engage strongback guides over air pallet.

L. Disengage crane hook, and with O-man engage seismic latch hook handle. Transport assembly into RML.

M. Unload strongback and canister from transporter in RML, with RML crane. (NOTE: Load may be approximately the maximum RML crane capacity.) Strongback has built in support feet and no base assembly is required in the RML. Transporter is then withdrawn from the RML so the shield door can be closed.

N. Removal and upending sequence is essentially the reverse of Steps A to M.

5.0 ESTIMATED WEIGHTS

5.1 The weight of system components is estimated as follows:

A.	Canister	- Empty	- 750 lb
		- Full (Volume Density Limit)	- 4350 lb
		- Full (Cask Design Limit)	- 2800 lb
		- Full (TMI Upender Limit)	- 2200 lb
B.	Core Debris	- Volume Density Limit	- 3600 lb
		- Cask Design Limit	- 2050 lb
		- TMI Handling Limit	- 1450 lb
		- Transporter Limit	- 3050 lb
C.	Strongback	- Empty	- 1200 lb
		- With TMI Limited Load	- 3400 lb

6.0 CANISTER DESIGN CRITERIA

6.1 It appears that only a few mandatory design criteria (Type A) for the canister design are imposed by the hot shop - RML handling equipment limits. There are, however, a number of desirable features which if included would be helpful and reduce costs of future handling equipment designs. (Type B)

6.2 Canister Criteria - Hot Shop and RML

Type A - Necessary
Type B - Desirable

ENGINEERING DESIGN FILE

PROJECT FILE NO. TMI-2
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FUNCTIONAL FILE NO.
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No.	Type	Criteria
1.	A	Closure design suitable for opening and reclosing with canister in horizontal attitude.
2.	A	Closure assembly provisions for attachment of remote handling equipment.
3.	A	Seals or gaskets on the closure joint should be replaceable, either remotely or on components which can be removed from the cell and easily decontaminated.
4.	A	Pilot guides, if a threaded closure is used, to guide initial thread engagement and to prevent damage to threads and seals during engagement.
5.	A	Provide a protected finished surface for seating of closure leak test fixture seals, as close as possible to the closure joint.
6.	B	Overall length tolerance $\pm$ 0.5 in. max.
7.	B	Side walls straight; i.e., constant cross section in region where lateral support is required.
8.	B	Closure assembly design to not restrict viewing from 90^o off axis of canister. (When in RML, window viewing limits preclude viewing from the "end" of the canister. Closure functional parts which were "recessed" would rely on less effective and less reliable TV viewing equipment.)
9.	B	Canister should have provisions for secure attachment of handling fixtures at the bottom end (opposite end from closure) for future handling.
10.	B	Torque reaction fittings at closure end of canister (as well as at bottom end, if that is TMI preferred location).

No.	Type	Criteria
11.	B	Canister design stressed for simple beam support at canister ends, for future handling.
12.	B	Closure designed for full support of a loaded canister resting on its (top) end in a fully inverted position.

7.0 STRONGBACK - PRECONCEPTUAL DESIGN CRITERIA

A. Use hot shop 10 ton or 100 ton hooks.

B. Use hot shop crane for actual laydown and upending. (Rotation should not require a drive mechanism.)

C. Use O-man or wall mount manipulators for latch, lock or adjustment functions as well as orientation of suspended loads.

D. System must also be reversible - to upend canister for pool rack storage.

E. Center of gravity of canister in strongback will vary with size and type of core debris load. Strongback shall accommodate any possible loading combination, including an empty canister.

F. Strongback shall be capable of a level suspended assembly in horizontal attitude with any possible CG.

G. Strongback shall engage the air pallet for seismic restraint during transport.

H. Strongback shall engage the air pallet with the load centered on the pallet deck and spread across the deck width.

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PROJECT/TASK TMI Fuel Canister Strongback

DATE November 12, 1982

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EDF PAGE NO OF

8.0 TASK REQUIREMENTS PACKAGE

APPENDIX E

STRUCTURAL CONSIDERATIONS FOR TMI-2 FUEL CANISTER STORAGE IN THE TAN 607 HOT SHOP POOL

EG&G Idaho, Inc.

FORM EG-A-...1831 (Rev 4-79)

ENGINEERING DESIGN FILE

PROJECT FILE NO

EDF SERIAL NO TMI-8

FUNCTIONAL FILE NO

PROJECT/TASK TMI Fuel Storage

DATE

SUBTASK

EDF PAGE NO ___ OF ___

SUBJECT Structural Considerations for TMI fuel canister storage in the TAN 607 Hot Shop Pool

ABSTRACT

At your request, we have reviewed the structural capability of the 30 year old pool floor for accepting a net storage load of 700 lbs/sq ft. In summary, while there is, from a practical standpoint, little doubt about the structure's ability to withstand this proposed loading, such an addition would not be technically acceptable according to the current design code. Following is a summary of our findings:

Analyses were made for structural compliance with American Concrete Institute (ACI) design requirements for both 1951 and 1977. Flexural stresses in the 14 inch thick concrete floor slab present no problem in either case.

However, shear stresses, while acceptable by 1951 standards, are excessive according to ACI 318-77 without assistance from an addition of shear reinforcement. As constructed, reinforcing steel which could be assumed to function as shear reinforcement consists of #6 bent bars at 14 inch centers. Their presence would, theoretically, be adequate except for current Code requirements which restrict the spacing of bent bar shear reinforcement to 8¼ inches, maximum, in order to better control possible cracking. In this respect, the proposed load addition would not meet technical requirements of the latest design code.

On the other hand, there are additional factors to be taken into consideration:

1. The probability of load-spreading

   Transverse reinforcement designed for the slab should be expected to function in this manner. Structural framework proposed for holding the fuel canisters should also be expected to assist in this regard.

   Acknowledgement of the probability for this mechanism to occur is found in ACI 318-77, Commentary 11.5.5, where it is stated, in part, "slabs

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J. S. Kilburn 12-3-82	A/E			UKlogen	12-3-82

EG&G Idaho, Inc.

FORM EG&G-2631A (Rev 4-78)

PROJECT FILE NO

ENGINEERING DESIGN FILE

EDF SERIAL NO TMI-8

FUNCTIONAL FILE NO

PROJECT TASK TMI Fuel Storage

DATE

SUBTASK

EDF PAGE NO 2 OF 2

ABSTRACT

are excluded (from the minimum shear reinforcement requirement) because there is a strong possibility of load sharing between weak and strong areas."

2. <u>The probability that existing concrete is now strong enough to withstand shear stresses without shear reinforcement</u>

Concrete can withstand a certain amount of shear without assistance from shear reinforcement. This shear strength is proportional to its compressive strength, which, under favorable circumstances, shows a remarkable increase with age. The storage pool slab has been exposed for many years to a continuous water cure, which is considered to be a most favorable environment for strength increase. For instance, test data for similarly treated concrete indicates the likelihood that the originally-required minimum 28-day compressive strength of 2500 psi increased to as much as 5000 psi after the first full year of water immersion. At any rate, after nearly 30 years of water curing, it's compressive strength may now be expected to be well over 6000 psi.

Current code requirements allow for the proposed 700 psf load addition without any shear reinforcement in the slab if the compressive strength of concrete is at least 5,540 psi.

As previously stated, this structure, as designed, does not meet the <u>letter</u> of the latest code. However, in view of the above factors, it appears that it does meet the <u>intent</u> of the code and should thus be adequate to perform its proposed function.

We trust this discussion will provide all the information needed. However, if you find that more definition or verification is required, we would be happy to assist with a program of core drilling and compression testing.

Concrete strength required for no shear reinforcing
(Concrete takes all the shear.)

11.3.1.1 $\rightarrow$ $\phi V_c = .85 \times 2 \sqrt{f'_c}\, bwd = V_u = 16.7^k$

$$\sqrt{f'_c} = \frac{16{,}700}{12 \times 11 \times .85 \times 2} = 74.4$$

$$f'_c = 5{,}538 \text{ psi}$$

"Design and Control of Concrete Mixes", 10th Ed., PCA, 1952

For concrete moist-cured after 28 days in air, at 28 days → 75% of strength of moist-cured concrete;

at 365 days → 150% " " "

Above situation was probable for Hot Shop Pool. Since 28 day test cylinders had to show at least 2500 psi, and since this pool concrete has been essentially moist-cured for 30 yrs., assume f'_c was 2500 psi in 28 days,

5000 psi in 365 days,

6000++ psi in 30 years.

Existing Concrete should now be much stronger than is needed for required shear stress.

Check with Ultimate Strength Design per 1963 requirements.

Ref. A.C.I. 318-63

$v = \frac{V_u}{bd} = \frac{16.7 \times 10^3}{12 \times 11} = 126$ psi

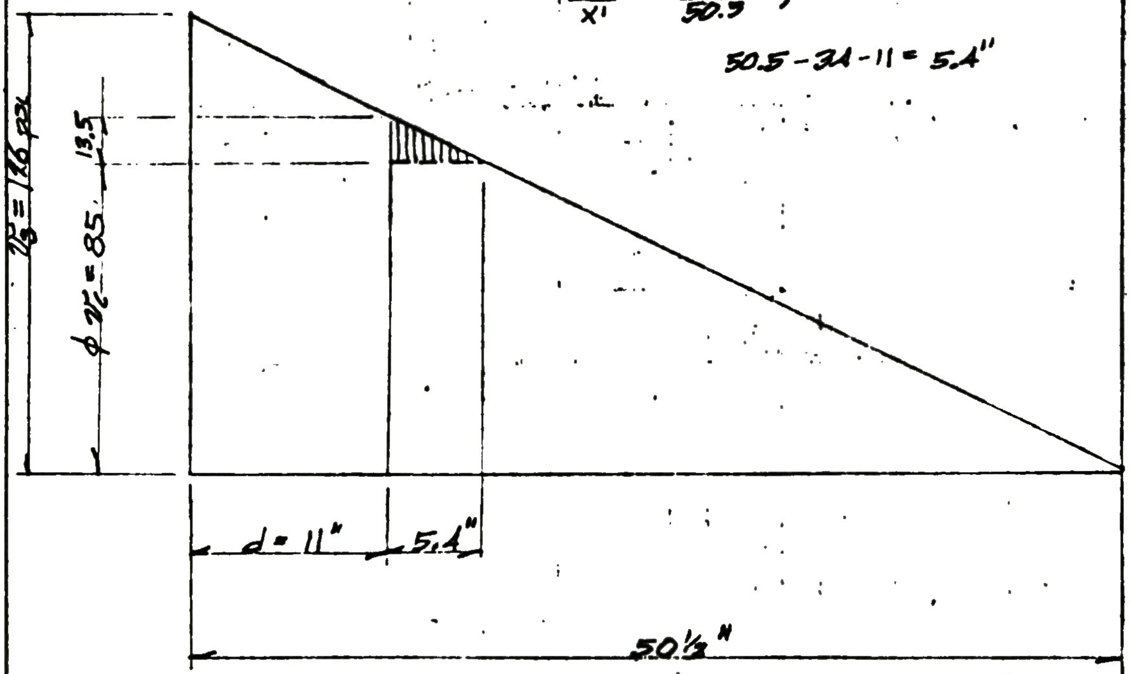
Max. v w/o shear reinforcement $= \phi 2 \sqrt{f'_c}$

$0.85 \times 2 \sqrt{2500} = 85$ psi

$\frac{X}{39.5} = \frac{126}{50.5}$; $x = 98.5$

$\frac{85}{X'} = \frac{126}{50.5}$; $x' = 34$

$50.5 - 34 - 11 = 5.4''$



As with sht. 2/3, need one bent bar @ 18" ctrs, except max. permissible spacing = $8\frac{1}{4}$" vs. 14" as-built

Ref. "Journal of the Am. Concrete Institute", April '51, pg 601

For beams with no web reinforcement, $v_c = 0.03 f'_c$ $= 150$ psi > 126 psi applied → No shear reinforcement req'd. for 1951 design

Shear (cont'd.) $V_u = 16.7^k$

$\phi V_c = 11.82^k$

$V_u - \phi V_c = 5.5^k < \phi 3\sqrt{f'_c}\, b_w d = 16.8^k$ OK

11-19 $V_s = A_v f_y \sin \alpha$

$= \frac{.72}{2} \times 40 \times .707 = 10.2^k$

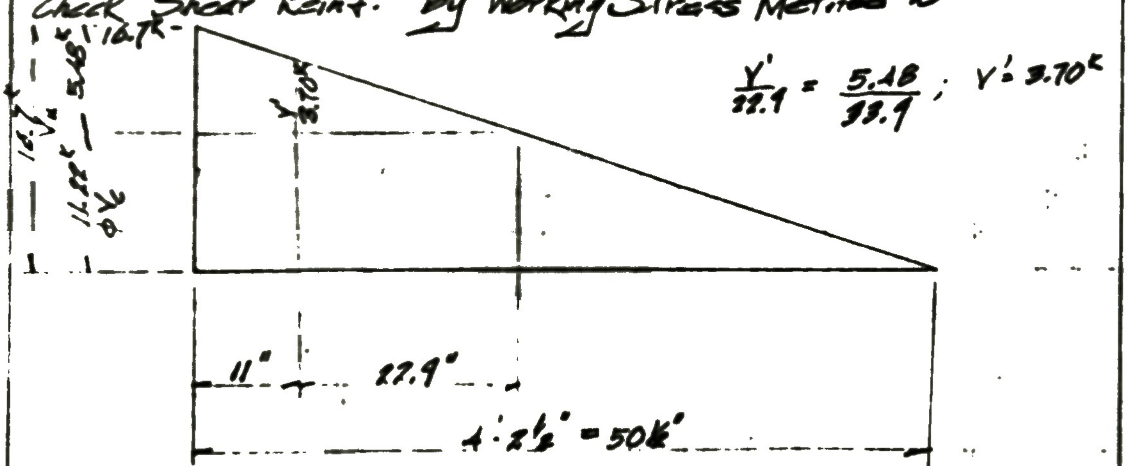
11.5.4.2 | Spacing Limits ~ $\frac{d}{2} = 5\frac{1}{2}" < 16"$ supplied

However, Commentary, 11.5.5 → Slabs excluded from min shear reinf. reqmts. because of load-sharing.

$V_n = V_c + V_s = 21.4^k > V_u = 16.7^k$

cf. "Reinf. Conc. Design Handbk, Working Stress Method, '65 pg. 39/Inclined Stirrups"

Check Shear Reinf. by Working Stress Method ~



$\frac{V'}{27.9} = \frac{5.48}{37.9}$; $V' = 3.70^k$

Diag. 17/ For $f_v = 20{,}000$ + #6 Inclined Bar, $A_v f_v = .44 \times 20{,}000 = 8{,}800$ ~ ½ fs, per 1301, '63 NS.

Table 16/ For $\alpha = 45°$, $B = 1.41$ $v' = \frac{V'}{bd} = \frac{3700}{12 \times 11} = 28$ psi

$\frac{v'b}{B A_v f_v} = \frac{28 \times 12}{1.41 \times 8{,}800} = .0271$

Table 16/ Max $s = d \times .75 = 8.25" < 16"$ supplied

Diag. 17/ Need 1 bent bar @ 18" ctrs > 8.25" max. allowable

Shear Reinf. OK except for Spacing

TAN Hot Shop Pool / Pool floor loading 2/5 11-29-82 JSK

Loads

21 ft/water ~ 21×62.4 = 1500 psf

Fuel @ 700 psf, net (Per Dave Wilkins) = 700

Ref. ACI 318-77 ~

pg. 29 / Req'd. str. $U = 1.4D + 1.7L$ | Design Lo:

$= 1.4 \times 1.5 + 1.7 \times .7 = 3.3$ ksf | $U = 3.3$ ksf

Commentary, pg. 40 ~

$$M_A = M_{allowable} = \phi M_n = \phi\left[A_s f_y \left(d - \frac{a}{2}\right)\right]$$

$$a = \frac{A_s f_y}{0.85 f'_c\, b} \qquad A_s = \#6 @ \frac{12}{2}" = .72\, in^2/ft$$

$$a = \frac{0.72 \times 40}{0.85 \times 2.5 \times 12} = 1.19$$

$$M_A = 0.90\left[.72 \times 40\left(11 - \frac{1.19}{2}\right)\right] = 269.6''^K = \qquad M_A = 22.5'$$

10.5.1 ~ $\rho_{min} = \frac{200}{f_y} = \frac{200}{40,000} = .005$

$\rho_{supplied} = \frac{A_s}{bd} = \frac{.72}{12 \times 11} = .00545 > .005$ req'd | ρ OK

Moment Supplied $= \frac{wL^2}{12} = 3.3 \times \frac{(8.417')^2}{12} = 19.5'^K < M_A$ | $M_{supplied} = 19.5'^K < M_A$ OK

check: AISC 2-126. Assume 3 equal spans / $M_{max} = .080\, wl^2 = 18.7^K$

$V_{max} = 0.60\, wl = 16.7^K$

check Shear ~ $\frac{8.417'}{2} \times 3.3 = 13.89^K < 16.7^K \rightarrow$ Use $V_u = 16.7^K$ | $V_u = 16.7^K$

$V_n = V_c + V_s$; $\phi V_c = \phi\, 2\sqrt{f'_c}\, b_w d = 0.85 \times 2\sqrt{2500} \times 12 \times 11 = 11.22^K$

TAN Hot Shop/Pool Floor Loading YS 12-1-82 JSK

Ref. Tech Specs., Invitation No. AT(10-1)-645 / 1-7-53

Div. 5-2, pg. 5 ~ Rebar per A305 & A15, Intermed. gr. new billet steel

Mesh per A185

pg. 6 F'c @ 28 days for Types I & II Cement = 2500 psi

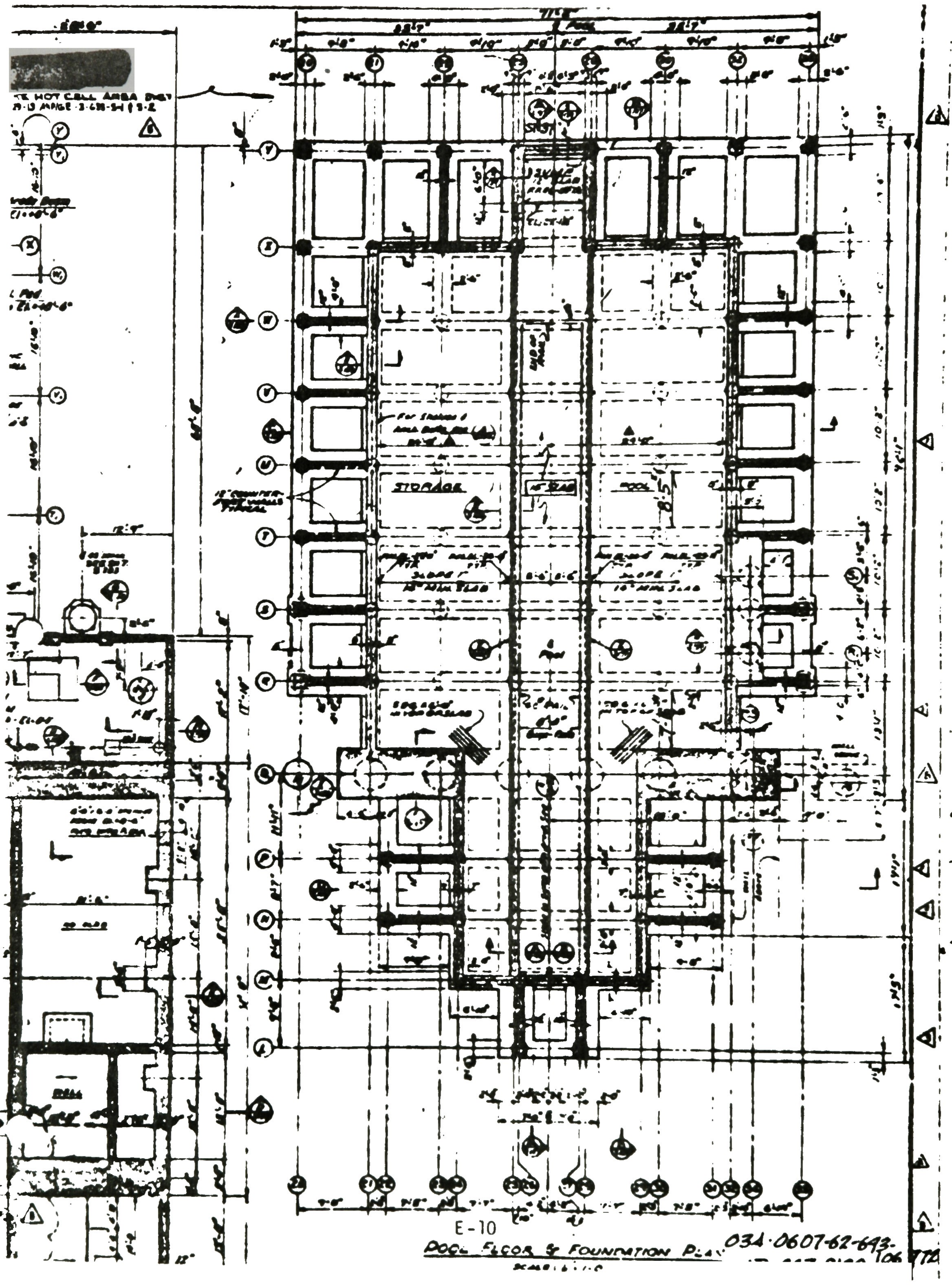
7 days for Type III = 2500 psi

F'c @ 7 days for I & II = 1500 psi

3 days for III = 1500 psi

pg. 27 Storage Pool concrete to include a set-retardant densifier in place of air entraining agent.

Div. 5-1, pg. 2 ~ Shafts for foundation piers shall be drilled a minimum of 2 ft. into rock, or as directed.



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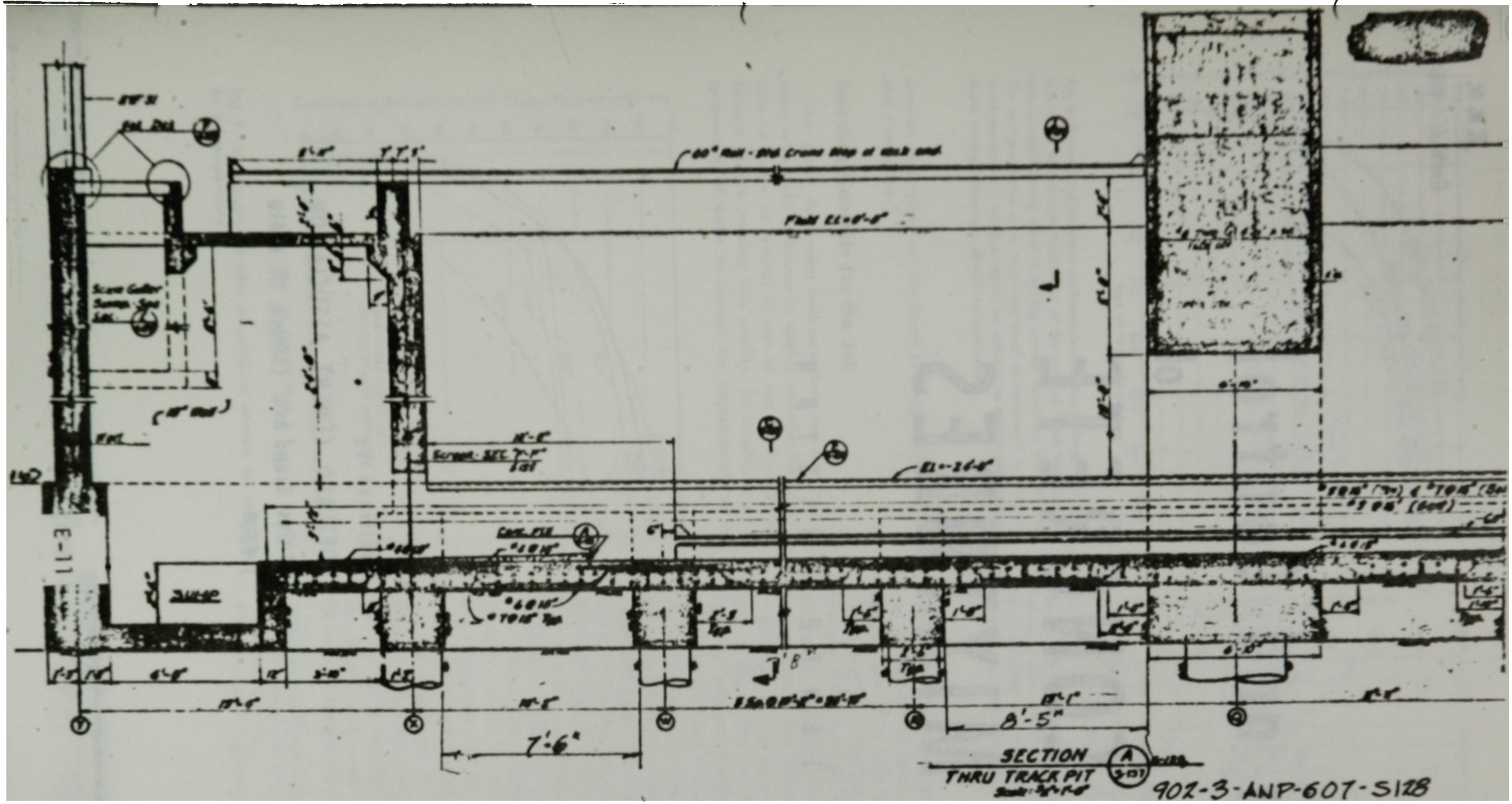

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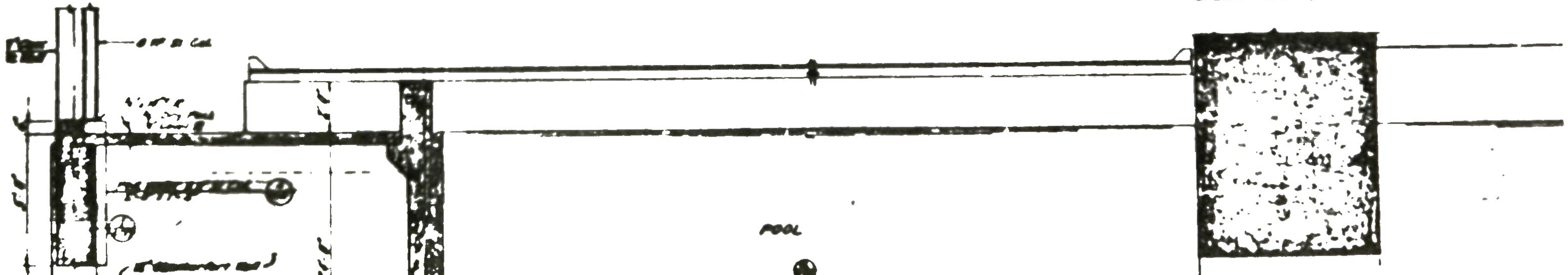

Design and Control of CONCRETE MIXTURES

TENTH EDITION

Published by
PORTLAND CEMENT ASSOCIATION
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1952

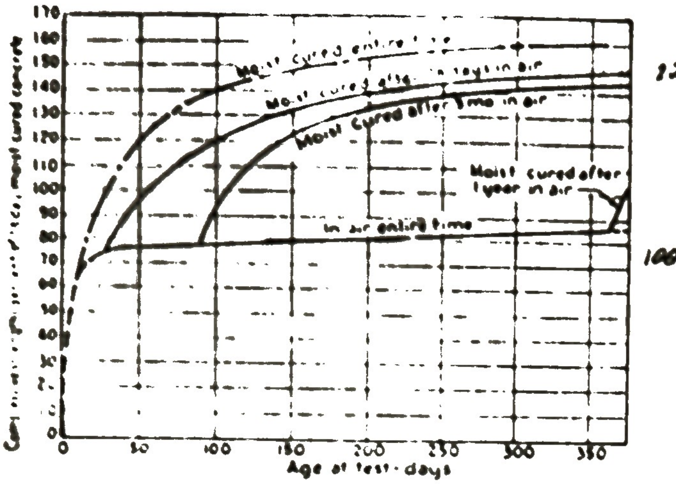

Fig. 8. Strength of concrete continues to increase as long as moisture is present to promote hydration of the cement. Note that resumption in moist curing after a drying period increases strength also. The test specimens were relatively small as compared to most concrete members in which it would be difficult to obtain resaturation. Moist curing, therefore, should be continuous.

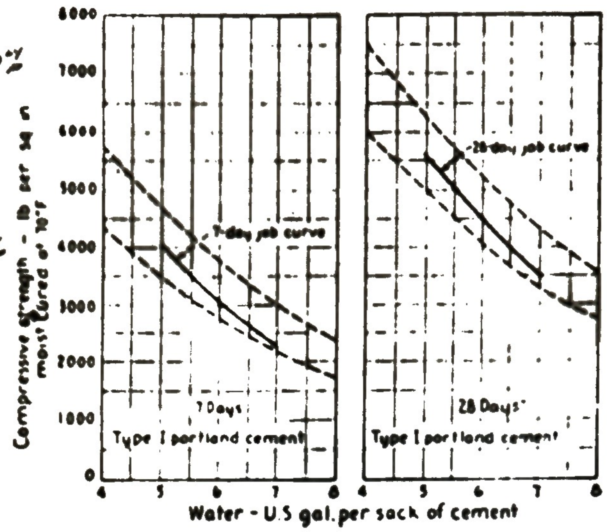

Fig. 10. On important work, tests should be made using the job materials from which job curves can be constructed. From these curves selection may be made of the water content which will produce the specified strengths.

weather concreting are discussed in Chapter 10, "Curing and Protection."

Selecting Concrete for the Job

The above discussion indicates that concrete can be made with wide variations in quality and that the proportion of water to cement is one of the most important factors in determining this quality. In establishing the proportion of water to cement the requirements of the finished structure must be considered. If it will be exposed to the elements or must be watertight, a water content must be selected which will produce a concrete resistant to the exposure conditions or which will be impervious to water. Table 1 can be used in making this selection. If strength is important the water content must be limited to the amount that will produce the desired strength.

Selection of this water content may be based on the data in Fig. 6 but, in the absence of any preliminary tests, the values indicated at the lower edge of the band should be used and a margin of safety should be allowed. The Joint Committee Report* recommends that the water content selected correspond to that required for strength 15 per cent higher than called for. Thus, if 28-day strength of 3000 psi is desired, the selection should be for 1.15x 3000 = 3450 psi. In Fig. 6 the water content indicated at the lower edge of the band for this strength at 28 days using Type I cement is 6¾ gal. per sack of cement.

For the utmost in economy and where relatively high strengths are to be used in design, tests for strength should be made with the materials to be used on the job and under job conditions. Such tests should follow standard procedures and should include not less than three different water contents. A job curve, such as shown in Fig. 10, can be developed from such tests and from this the proper water content can be selected. Here also the 15 per cent margin should be allowed.**

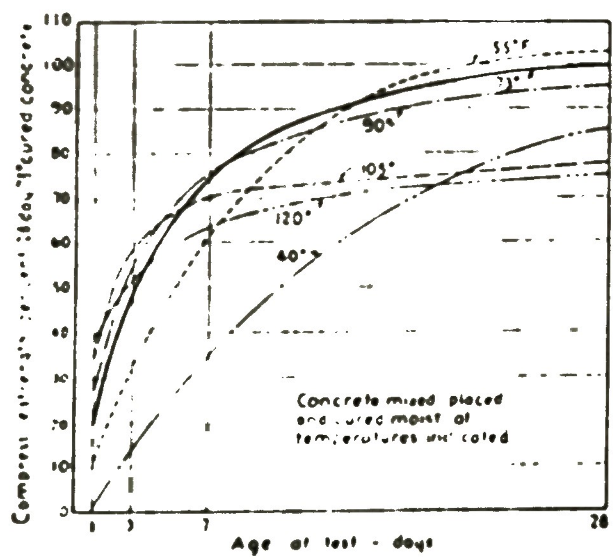

Fig. 9. The strength and other properties of concrete are affected by the temperature.

*Recommended Practice and Standard Specifications for Concrete and Reinforced Concrete, June 1940, available from American Concrete Institute.

**See Method 2 of Specifications, page 49.