Nuclear

GPU Nuclear Corporation
Post Office Box 480
Route 441 South
Middletown, Pennsylvania 17057-0191
717 944-7621
TELEX 84-2386
Writer's Direct Dial Number:

(717) 948-8461

4410-85-L-0181
Document ID 0316A

October 3, 1985

TMI Program Office
Attn: Dr. B. J. Snyder
Program Director
US Nuclear Regulatory Commission
Washington, DC 20555

Dear Dr. Snyder:

Three Mile Island Nuclear Station, Unit 2 (TMI-2)
Operating License No. DPR-73
Docket No. 50-320
Early Defueling Safety Evaluation Report
Response to Comments

Attached are responses to your comments on the Early Defueling Safety Evaluation Report (SER) transmitted by your letters dated July 29, 1985, and August 13, 1985. Note that additional input is being prepared to respond to your question concerning onsite testing of canisters and will be submitted under separate cover.

Sincerely,

F. R. Standerfer

F. R. Standerfer
Vice President/Director, TMI-2

FRS/RBS/eml

Attachments

cc: Deputy Program Director - TMI Program Office, Dr. W. D. Travers

D009
1/1

8510080457 851003
PDR ADOCK 05000320
P PDR

1. Comment: Submit Revision 2 to the Boron Dilution Hazards Analysis and the Safety Evaluation Report for Heavy Load Handling.

Response: The Safety Evaluation Report for Heavy Load Handling was submitted to the NRC on September 11, 1985, GPU letter 4410-85-L-0172, Standerfer to Snyder. The Boron Dilution Report was submitted to the NRC on September 27, 1985, GPU letter 4410-85-L-0195.

2. Comment: Describe the procedural and physical controls provided to prevent the inadvertent lifting of debris out of water during defueling activities. Discuss the feasibility of providing mechanical means to restrict the lifting of long-handled tools. What is the highest level that debris could reach if lifted with any combination of rigging bar and long-handled tool?

Response: Prevention of inadvertent lifting of debris out of the water during planned defueling activities will be generally implemented mechanically through the use of rigging bars. It is planned to have three standard rigging bar lengths. The different length bars may be combined in several different ways so as to accommodate the different rigging bar length requirements for the various defueling tools. The effective use of the rigging bars will require adherence to administrative controls on the rigging of tools. Using the rigging bars per the administrative controls will prevent debris from being lifted above elevation 323'-6". The lengths of the separate rigging bars are such that improper combinations of the separate bars (i.e., failure to comply with administrative controls) could result in debris being lifted out of the water and even onto the work platform.

The above approach only applies to those tools handled by the reactor building service crane. For tools hanging from the work platform jib cranes, no rigging bars are required, as the vertical movement limitations of the jib cranes prevent the lifting of debris above elevation 323'-6".

Administrative controls will be employed for lifting debris above elevation 323'-6". The governing criteria will be the maintenance of acceptable radiation levels on the defueling work platform. In addition to the design features and administrative controls, radiation monitoring capability will be provided on the defueling work platform to alert the operators to dose rates that may be unacceptable. Acceptable dose rates will be established by Radiological Controls for the planned activities.

It is not considered advisable to provide mechanical means to restrict the lifting of long handled tools. This is because the governing criteria is to maintain acceptable dose rates on the defueling work platform, not the depth of water over the debris being handled. To mechanically restrict the lifting of debris could result in unnecessary cutting and handling activities which would result in unnecessary occupational exposure. The planned radiation monitoring capability along with the planned physical and procedural controls are considered adequate for maintaining acceptable dose rates on the defueling work platform.

3. Comment: Discuss the precautions that will be taken during early defueling to prevent damage to any incore instrumentation strings.

Response: During early defueling activities, structurally intact in-core instrument strings are not expected to be found above the "hard stop" in the core region. The precaution to be taken if one is found is to cut it or handle it carefully. This precaution will be incorporated into the GPUN procedure for Defueling Operations. This precaution will minimize the possibility of transmitting loads to the weld between the in-core instrument nozzles and the reactor vessel lower head.

4. Comment: Describe the equipment used and the methods for weighing the three types of canisters during defueling operations and following dewatering in the fuel handling building.

Response: Knockout Canisters - Knockout canisters are weighed during all vacuuming operations using the knockout canister connect assembly module, which is illustrated in Figure 1. The knockout canister connect assembly module is mounted to the underside of the shielded work platform. After a mechanical interlock between the Canister Positioning System (CPS), which is carrying the knockout canister, and the knockout canister connect assembly module has been established, the knockout canister is lifted from the CPS a predetermined distance (approximately 1 inch) to allow weighing of the canister. The existing load cell (0-7500 lb. range, $\pm$ 0.5% accuracy) sends a signal to the Control Console on a continuous basis that reflects the weight of the knockout canister. A load cell readout is provided on the Control Console.

Filter Canisters - Filter canisters are weighed continuously during all vacuuming operations by the Filter Canister Weighing System. The Filter Canister Weighing System, illustrated in Figure 2, will be attached to the filter canister and suspended from the shielded work platform deck shielding plate. The load cell will provide the means for constantly monitoring the weight of the filter canister. The existing load cell has a range from 0-7500 lb., $\pm$ 0.5% which results in a $\pm$ 17 lb. accuracy for the weight of a loaded canister (3355 lb). The load cell readout is located on the Control Console.

The location of both filter canister weighing systems is shown in Figure 3.

Fuel Canisters - Fuel canisters are weighed on an as-needed basis using the B&W Canister Grapple Tool (open fuel can lifting tool) and a typical weight scale in the rigging of a crane (See figure 4). Since fuel canisters are open topped during loading, visibility during loading permits the operators to fairly accurately judge the "fullness" of the canister, and thus only weigh the canister once or twice for verification.

Following dewatering (in either the reactor building or the FHB) each canister is picked up and raised out of the water by the canister handling trolley. This allows the load cell in the canister handling trolley lifting path to sense the canister out-of-water weight to within ± 35 pounds.

5. Comment: Describe RCS sampling capability, frequency, and sampling locations during early defueling.

Response: The frequency of RCS sampling is defined by the Hazards Analysis: Potential for Boron Dilution of the Reactor Coolant System. The primary location for RCS sampling is the already installed RCS sample pump. Secondary locations will be:

- o upstream and downstream of the reactor vessel filter - DWCS
- o upstream and downstream of the ion exchanger - DWCS
- o recirculation sample from the bleed tanks
- o grab sample from the reactor vessel
- o on line boronmeter from reactor vessel

6. Comment: What conditions could require flooding of the fuel transfer canal (FTC)? To what level would the FTC be reflooded to recover from an abnormal event? Is all water sensitive equipment above the maximum reflood water level?

Response: Canal flooding could be required in the event of excessive leakage from either the canal dam or the IIF. In this situation the lowered water level could result in higher radiation levels.

The highest water level to which the fuel transfer canal/refueling pool would be flooded in recovery from postulated abnormal events is 327'6", which is the approximate normal defueling water level. This will provide adequate water shielding of the plenum, canisters and core debris. The water sensitive defueling equipment electrical components are located above this elevation.

7. Comment: Describe the design and operation of the canister retention mechanisms on the bottom of the canister transfer shield. What controls will be implemented to restrict horizontal movement of canisters? Provide drawings of the canister handling bridges, canister transfer shields and related components.

Response: The bottom of the canister transfer shield houses the canister keepers. In the unlikely event that the grapple drops a canister, these devices will absorb the impact energy of the dropped canister and physically restrain it from further vertical motion. They position themselves for operation automatically after the canister has been lifted into the CTS and the shield collar retracted to its full-up position; they automatically retract as the shield collar is lowered past the bottom edge of the fixed CTS. The canister keepers are shown in Figures 5 and 6. The detailed design of the canister keepers is shown on NES drawings 83E1614 and 83E1615 (See Table 2).

Plainville Electric Products Co. Drawing 85013 Sheet 17 (see Table 2) shows the location of operator controls on the motor control cabinet front door. Drawing 85013 Sheet 18 shows the location of the digital displays which are mounted directly above the door and provide the operator with readouts of hoist load, and collar, bridge and trolley positions. Control of the trolleys is accomplished by the use of a Programmable Logic Controller (PLC). The controller has been programmed to provide numerous safety interlocks to guard against accident conditions such as dropping and exposing a fuel filled canister. These interlocks may be bypassed under controlled conditions by use of key operated switches mounted on the control panel.

There are five operator control functions:

1. Hoist - Move Up or Down
2. Trolley - Move East or West
3. Bridge - Move North or South
4. Grapple - Engage Canister or Disengage Canister
5. Collar - Move Up or Down

Each "Move" function is allowed to occur only within a "Travel Zone".

The permitted "Travel Zones" are monitored by encoders mounted on the Hoist, Trolley, Bridge and Collar.

In addition to encoders, limit switches have been provided to prevent bridge and trolley from going beyond maximum travel limits in the event of an encoder failure. Limit switches are also provided on the hoist to prevent contact of the grapple support plate with the hoist sheaves, and on the collar to detect contact when it is being lowered. During normal operation the operator must remain within the "Travel Zones" and is prevented from going beyond the limits by the programmable controller. "Override" key switches may be used to allow travel outside the travel zones. The maximum travel limits defined by limit switches cannot be overridden.

In addition to travel limits, there are additional movement restrictions. These movement restrictions may be referred to as "Operation Sequence Limits". An example of an Operation Sequence Limit (OSL) is that no movement of the bridge or trolley is permitted if the collar is in the lowered position. "Override" key switches may be used to allow movement for many OSL's. No override is provided to permit more than one command to be in effect at one time. Also, only one override may be in effect at one time.

Whenever a "Travel Zone" limit is reached an indicating light, "Area Travel Limit" or "Up Down Travel Limit", will come on. Whenever an "Operation Sequence Limit" prevents motion requested by the operator, an indicating light, "Operator Error", will come on. Table 1 summarizes the control logic.

Table 2 provides a list of drawings for canister handling bridges, canister transfer shield, and related components, which were previously provided to the NRC.

8. Comment: Identify all materials with the potential to affect RCS reactivity during defueling. The administrative limit on boron concentration (4950 ppm) is based on maintaining a level that will allow detection and correction of a boron dilution event prior to reaching the Technical Specification limit of 4350 ppm. Since the introduction of materials that may increase reactivity would reduce this margin of safety between the administrative and T. S. values, what is your justification for using the current administrative limit in the criticality analysis addressing the introduction of these materials into the RCS? Discuss means to prevent the inadvertent introduction of these materials into the RCS and specify limits on the quantities that could be introduced.

Response: It is correct that an administrative limit of 4950 ppm is used to provide a margin for mitigating a boron dilution event prior to reaching the Tech Spec limit of 4350 ppm. The primary consideration in setting this limit was the inadvertent addition of unborated (or underborated) fluid into the reactor vessel which would mix with the borated water and reduce the overall RCS boron concentration (i.e., a boron dilution). Underborated water was the major concern in this regard. A separate concern is the introduction of foreign materials into the vessel which would not mix with the borated water, but would instead, remain intact. As discussed in the following paragraphs, such materials could act as reflectors or moderators. The quantities of non-miscible materials are limited in the area of the reactor vessel to assure an adequate margin over criticality, assuming an initial boron concentration of 4950 ppm, if they should inadvertently enter the vessel (e.g., spillage of fluids on the work platform). The administrative limit rather than the Tech Spec limit is used as a basis for criticality calculations involving the introduction of non-miscible materials into the reactor vessel because the administrative limit is the lower bound at which the RCS boron concentration is maintained by plant procedure. (The actual boron concentration has been significantly higher since the 4950 ppm administrative limit was set). The only mechanism for reducing the overall RCS boron concentration below the 4950 ppm administrative limit is a boron dilution event. The probability of a simultaneous dilution event and the introduction of non-miscible materials into the vessel is considered negligible. No scenario has been identified that results in both a system boron dilution event coupled with an introduction of non-miscible materials into the reactor vessel in a single credible accident or maloperation. Thus, the administrative limit is an appropriate assumed initial condition for criticality calculations involving the introduction of non-miscible materials into the vessel.

The limits on the amount of foreign materials that can be introduced into the RCS and still maintain $k_{eff} \leq 0.99$ have been established. Any material introduced into the RCS has the potential to affect the RCS reactivity. Practically, it was considered impossible to identify and analyze every material that could possibly be introduced, consequently a bounding analysis approach was used. The general approach in the development of these limits was to group the foreign materials into two material types - reflecting materials or moderating materials. The materials that fell into the reflecting material category were those materials that cannot become interstitially dispersed within the fuel (e.g. structural steel, lead shielding and plastic tubing). As initial analyses indicated large amounts of reflecting materials could be introduced without significantly increasing k_{eff}, not all the potential reflecting materials were specifically analyzed. Instead a representative, yet conservative, list of material was developed for a series of criticality analyses. This list included unborated water, beryllium, lead, iron, carbon and polyethylene. It should be noted that the inclusion of a material on this list does not indicate the possibility of the material being introduced into the RCS. Rather, the list was developed to provide conservative reactivity effects associated with the introduction of foreign reflecting materials. Additionally, it should be noted that for the purposes of the reflecting material analyses, unborated water was treated as a solid material, incapable of intermixing with the fuel. Unborated water was also treated as a moderating material (i.e., capable of becoming interstitially dispersed within the fuel) as discussed below. The results of the analyses for the specified materials were used in the conclusion that there will be no limits on the amount of foreign reflecting materials that can be introduced either accidentally or intentionally into the RCS.

The materials that were classified as moderating materials were those materials capable of becoming interstitially dispersed within the fuel (e.g., liquids and greases). Moderating materials that have been identified include the hydraulic fluid in the defueling tooling hydraulic system, lubricating materials for the various defueling equipment, as well as the working fluids for the core bore equipment. More materials may be identified as the defueling procedures become better defined. As with the reflecting materials, only a selected list of moderating materials were specifically analyzed. This list consisted of unborated water and the originally proposed defueling tooling system hydraulic fluid. The results of the analyses for these fluids were used to establish a two (2) gallon limit on the amount of underborated (i.e., < 4350 ppm) foreign moderating materials allowed to be intermixed with the fuel. As the defueling tooling hydraulic fluid will now be borated to a minimum of 4350 ppm, the two gallon limit does not apply to this fluid.

The general approach to prevent inadvertant introduction of unacceptable amounts of foreign materials will be to administratively control the amounts of materials handled above or within the reactor vessel.

9. Comment: Describe the number, type, and location of radiation monitors to be used during early defueling.

Response: Radiological Engineering through the radiological review process will initially require a 'Victoreen Vamp' or equivalent alarming dose rate meter be placed in each of the following locations when in use:

- The defueling rotating platform as close as practical to the slot
- The fixed platform in the control station area
- On each of the Reactor Building and Fuel Handling Building Canister Handling Bridges
- On the Fuel Handling Building Dewatering Platform

Setpoints are expected to be about double the normal ambient radiation fields. In addition an 'AMS-3' or equivalent will be installed on the fixed platform to provide continuous air beta gamma particulate monitoring. As experience with the defueling system is gained, Radiological Engineering may change numbers, locations, and setpoints of instrumentation installed by Radiological Controls as need or cause arises.

Neutron monitors for criticality monitoring per 10 CFR 70.24 will be installed at three locations:

- o The Reactor Building at about the 347' elevation at the south end of the refueling canal
- o At the 'A' Fuel Pool at about the 349' elevation at the north end of the pool
- o In the truck bay at about the 315'elevation

The alarm setpoints and sensitivity will be able to detect radiation levels about two orders of magnitude below that required by the regulation. More detail is provided in the Recovery Ops Plan Change Request No. 34.

10. Comment: Provide revised occupational exposure estimates for early defueling when available.

Response: Revised man-hours are not yet available and therefore man-rem revisions are also not available. Until more training and even experience with actual defueling activities is gained, revision of man-hour estimates would not provide any more accuracy that already exists. The estimates included in the table were developed in June of 1984. No provision was included to allow separation of 'early' and 'bulk' defueling activities. This detail can only be provided after new estimates of man-hours for various tasks are available.

The man-rem numbers provided in the SER are current best available estimates but should not be viewed as an absolute estimate. The number used was originally developed to provide comparison between several defueling system design options and has never been intended to be Radiological Controls' exact estimate of defueling man-rem. These numbers may not include items which were common to all of the options which were under study. The estimate may be considered to be a fairly reasonable 'first pass' estimate and should be considered as providing an indication of the magnitude of the expected doses, not an exact estimate.

11. Comment: Describe all planned RCS processing activities prior to defueling. Will RCS concentrations of Cs and Sb be reduced to the levels referenced in the SER as a prerequisite for the commencement of defueling?

Response: RCS processing is planned before the start of defueling to reduce the antimony concentrations in the RCS water. After the initial processing, continued processing with the modified DWCS Reactor Vessel portion will continue with a slip stream directed through SDS with make-up from processed RCS.

The SER for Early Defueling of the TMI-2 Reactor Vessel will be revised to reflect the projected RCS concentrations of Cs, Sb and Co. The RCS water will be processed to limit the exposure to personnel on the work platform to within acceptable dose rates. However, the RCS concentrations of these radioactive materials do not have to be reduced to the levels projected to start defueling.

12. Comment: In the event of a canister drop over the dry canal, what are the worst case dose consequences to workers from both direct exposure and airborne contamination?

Response: As was discussed in the Early Defueling SER, the canister transfer shield (CTS) has been designed with diverse means for preventing a canister drop from occurring during canister transport. Thus, for a canister drop to occur over the dry portion of the refueling canal, multiple failures must occur. Consequently such a drop is considered highly unlikely.

To provide the worst case direct exposure dose to personnel resulting from a canister drop over the dry canal no credit was taken for the shielding provided by the CTS. This approach is quite conservative since, by design, if a canister were to drop into the dry canal it could not fall completely out of the CTS.

The maximum resulting dose rate at a location 5 feet from the unshielded canister was calculated to be 46 rem/hr. At a location 11 feet from the unshielded canister, the maximum dose rate was calculated at 13 rem/hr.

In addition to the unlikeliness of dropping the canister, potential airborne contamination resulting from canister leakage is limited by the following features:

- o By design the lift height of the canister is such that the canister will not fall completely out of the transfer shield if a drop were to occur over the dry portion of the canal. This ensures any impact will occur on the canister bottom head.
- o Vertical drop tests have shown that the bottom head of the defueling canisters can withstand an impact energy of 51,300 ft-lbs with minor deformation and no observed cracking. This corresponds to an impact velocity of 34 ft/sec on an unyielding surface. This impact load exceeds the calculated impact load for a canister drop in the reactor building. Therefore, the bottom head of the canister would not be expected to crack or rupture.
- o Limited space is available for leakage of canister contents due to the small inner diameter of the canister transfer shield. The maximum annular space width is estimated at 1/2 inch. The small clearance between the

canister and the shield will provide structural support along the length of the canister and prevent a total circumferential rupture of a canister; therefore, leakage would be expected to occur only at the extreme ends of the canister.

- o The top portion of each canister contains the most likely leakage path. Under normal conditions, the canister vent and drain connections on the upper head may offer a leakage path from the canister during connect/disconnect operations. These connections, Hansen quick disconnects, have integral shutoff valves and will be capped before shipping. If leakage should occur, it is expected to consist of fuel fines, gases, and water vapor.

- o The upper closure head nozzles on the canisters are protected by a steel skirt. Under postulated drop accidents, direct impact loads on the canister can be minimized. There is no defined mechanism for dropping something inside the skirt which would directly impact the nozzles. Therefore, leakage from the canister due to a direct impact on the nozzles is not credible.

These design features of the canister and the handling equipment make the potential for a leak very small. It is expected that under the postulated dry canal drop conditions no leakage from the canister will occur. Therefore, any air contamination doses would be minimal.

13. Comment: Provide your evalution showing that potential accident consequences and occupational exposures that may result from early defueling activities are bounded by the events analyzed in the PEIS.

Response: A section titled "Environmental Assessment" will be added in the next revision of the Early Defueling SER. This section will address the environmental impact of the planned early defueling activities.

14. Comment: Discuss the testing for design verification that was done during the design and selection process for defueling canisters. Discuss your program for assuring that canisters delivered from your vendors will meet the design requirements. Describe the functional testing and checkout that will be performed on each canister prior to its use for defueling.

Response: Design Verification Testing

Prior to full scale production of filter bundles, single element filter flow tests and dewatering tests were conducted. These tests used single filter modules in a housing which simulated a canister shell. Water with suspended particles simulating core debris was pumped through the module at the actual flow rate expected during use. The tests were successful in demonstrating the adequacy of the filter media, both in filtering capability and dewatering capability. The modules were successfully dewatered after loading with debris by a gravity drain method. This is conservative since actual dewatering is driven by a differential pressure of 1 psid for filters.

Prior to production of knockout canisters, the design of the internals arrangement was tested in a full scale mockup at full flow. The design was adequately verified in this test since the rate at which particles were removed by the test was adequate for defueling operations. The dewatering capability of the knockout canister was demonstrated in a full scale mockup of the dewatering piping internal to the canisters. The test included a section of production "rigimesh" drain screen and simulated debris. Dewatering was accomplished using a 1 psid differential pressure. The test showed that the canister design afforded effective dewatering capability. The "rigimesh" filter was unaffected by the debris and the effluent was clear. This test adequately demonstrated dewatering capability in the fuel canister also since the drain screen and piping is the same. Note that the dewatering tests are conservative since the differential pressure which will actually be used is higher than 1 psid.

Other testing has also been performed to verify the design of the canisters as follows:

- o Drop testing of fuel canister
- o Lift testing of canister grapple socket and open fuel canister sockets
- o Checkout testing of dewatering tools and quick disconnect fittings
- o Tensile test of filter media
- o Prototype testing of B_4C poison pellets
- o Static ultimate strength test of knockout canister support spiders
- o Recombiner catalyst functional tests

Canister Fabrication

Having verified the design of the canisters as stated above, the canisters are then fabricated in accordance with the design documents. Canister fabricators are required to perform all work to a quality program which meets the requirements of ANSI N45.2. Implementation of the program is being verified by GPUN. Each completed product is verified by the fabricator to be correct and complete with regard to components, materials, and dimensions. Additional information will be provided by separate letter.

TABLE 1: TMI-2CANISTER HANDLING TROLLEY CONTROL LOGIC

FUNCTION	REQUIREMENTS	OVERRIDE
Enable Controls	Enable Control KS on	
Command	Only One Command Allowed at a Time	
Override	Only One Override at a Time	
Horn	Weight Overload	
Permit Grapple Release	Grapple Contact	or Grapple Disengage
Permit Bridge North	Bridge Not North Limit Switch and Bridge Not North Travel Zone Limit and Collar Withdrawn and No Motor Overload	or Bridge/Trolley Travel Limit or Bridge/Trolley Move
Permit Bridge South	Bridge Not South Limit Switch and Bridge Not South Travel Zone Limit and Collar Wtihdrawn and No Motor Overload	or Bridge/Trolley Travel Limit or Bridge/Trolley Move
Permit Bridge East	Trolley Not East Limit Switch and Trolley Not East Travel Zone Limit and Collar Wtihdrawn and No Motor Overload	or Bridge/Trolley Travel Limit or Bridge/Trolley Move
Permit Bridge West	Trolley Not West Limit Switch and Trolley Not West Travel Zone Limit and Collar Wtihdrawn and No Motor Overload	or Bridge/Trolley Travel Limit or Bridge/Trolley Move

TABLE 1: (Continued)

FUNCTION	REQUIREMENTS	OVERRIDE
Bridge Slow Speed	Bridge Within 8 Inches of Maximum Travel	
Trolley Slow Speed	Trolley Within 8 Inches of Maximum Travel	
Permit Lower Collar	Collar Not Extended and No Platform Contact and No Motor Overload	or Collar Up
Permit Raise Collar	Collar Not Withdrawn and Hoist Withdrawn and No Motor Overload	or Collar Up or Collar Up
Collar Slow Speed,	Collar Lower Command and Over Platform	
Permit Raise Hoist	Hoist Not Withdrawn and Grapple Open or Grapple Closed and Collar Extended or Grapple Not Loaded and No Weight Overload and Keeper Disengaged and No Motor Overload and Hoist Not Up Limit Switch	or Hoist Up/Down or Hoist Up/Down or Canister Hoist Raise
Hoist Slow Speed Raise	Fuel Handling Building Hoist < 119" or Hoist > 300" or Hoist > 212" in dewatering area Reactor Building Hoist < 24" or Hoist > 271" or Hoist > 237" over work platform	

TABLE 1: (Continued)

FUNCTION	REQUIREMENTS	OVERRIDE
Permit Lower Hoist	Hoist Not Extended (367" in R.B., 365" in F.H.B.)	or Hoist Up/Down
	and Grapple Open or Grapple Closed	or Hoist Up/Down
	and Hoist Cable Not Slack	or Hoist Up/Down
	and Collar Extended or Grapple Not Loaded	or Canister Hoist/Raise
	and Keeper Disengaged	
	and No Motor Overload	
Hoist Slow Speed Lower	Fuel Handling Building Hoist > 344" Reactor Building Hoist > 271" or Hoist > 237" over work platform	

Table 2 - Drawing List

Supplier Drawing No.	Revision	Title	Bechtel Dwg. No.
83E1599	2	Canister Handling Trolleys Outline Dimensions	15737-2-M067-00102-03 Transmittal 01150
83E1600	1	RB Canister Handling Trolley General Arrgt.	15737-2-M067-00157-02 Transmittal 01128
83E1601	1	FHB Canister Handling Trolley General Arrgt.	15737-2-M067-00158-02 Transmittal 01128
83E1602	3	Canister Transfer Shield	15737-2-M067-00031-04 Transmittal 01128
83E1614		Details - Sheet 6	15737-2-M067-00043-02 Transmittal 01208
83E1615	1	Canister Keeper	15737-2-M067-00044-02 Transmittal 01208
9729-1 Sht. 23	D	Bridge and Trolley Assembly	15737-2-M067-00154-04 Transmittal 01209
85013 Sht. 17		Switches and Pilot Lights	15737-2-M067-00059-04 Transmittal 01134
85013 Sh. 18	2	Display Layout	15737-2-M067-00092-02 Transmittal 01134

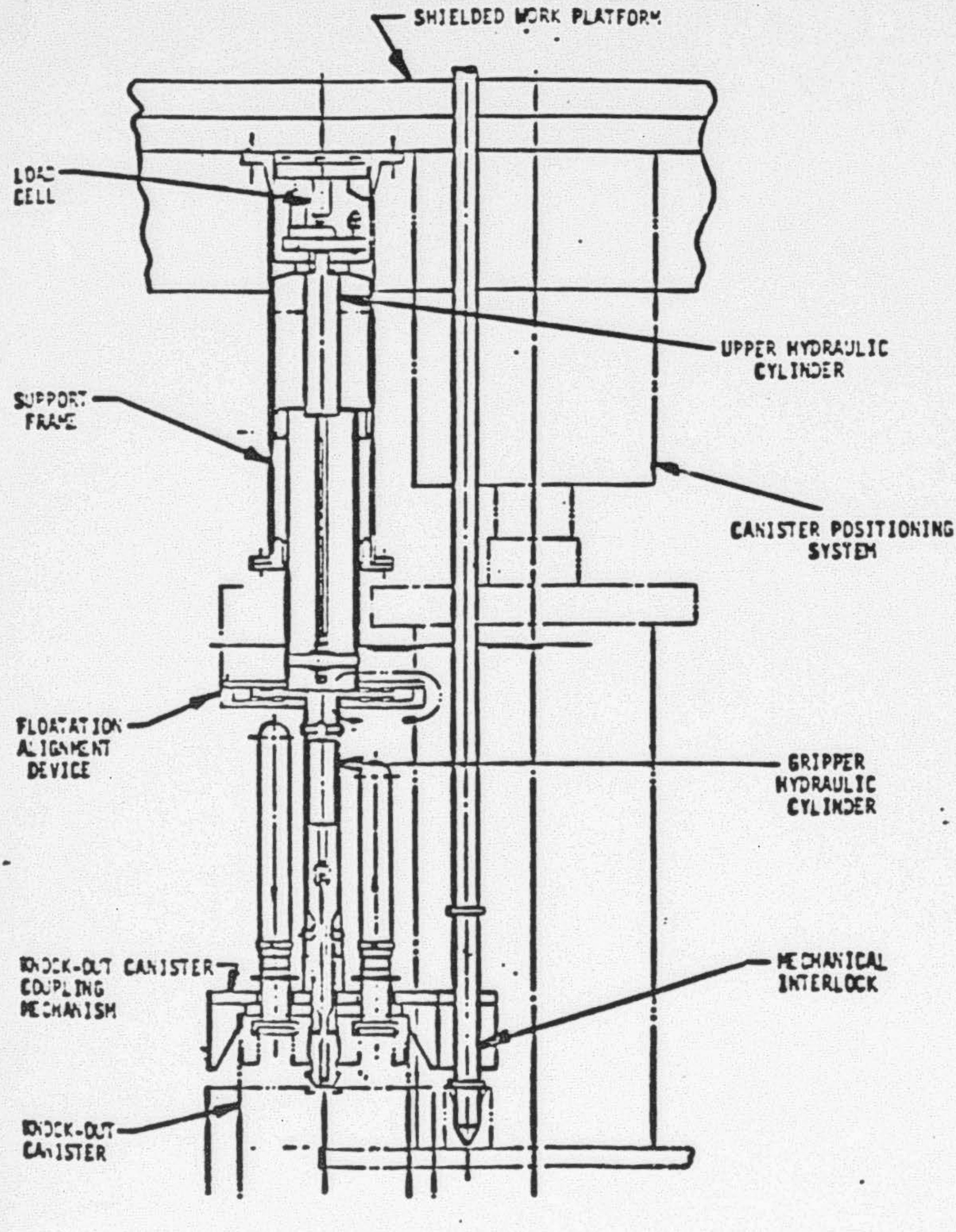

Figure 1: Knockout Canister Connect Assembly

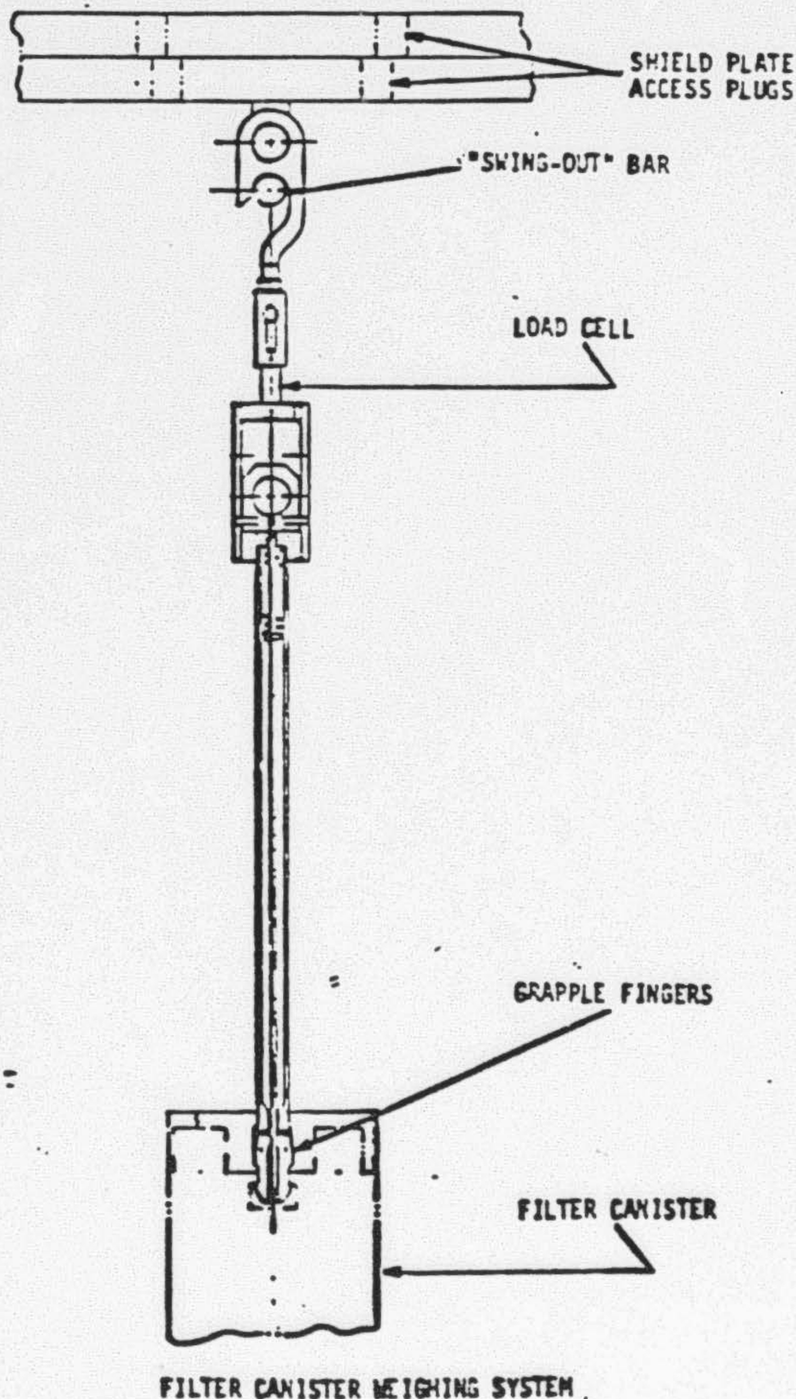

Figure 2: FILTER CANISTER WEIGHING SYSTEM

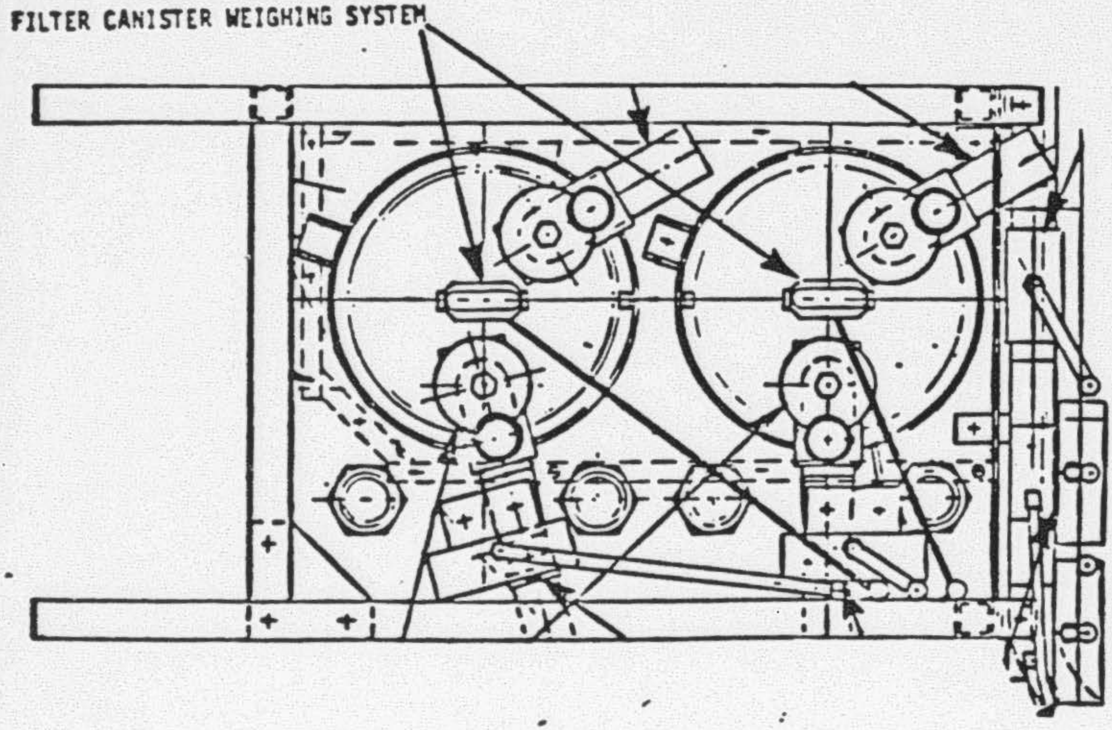

Figure 3

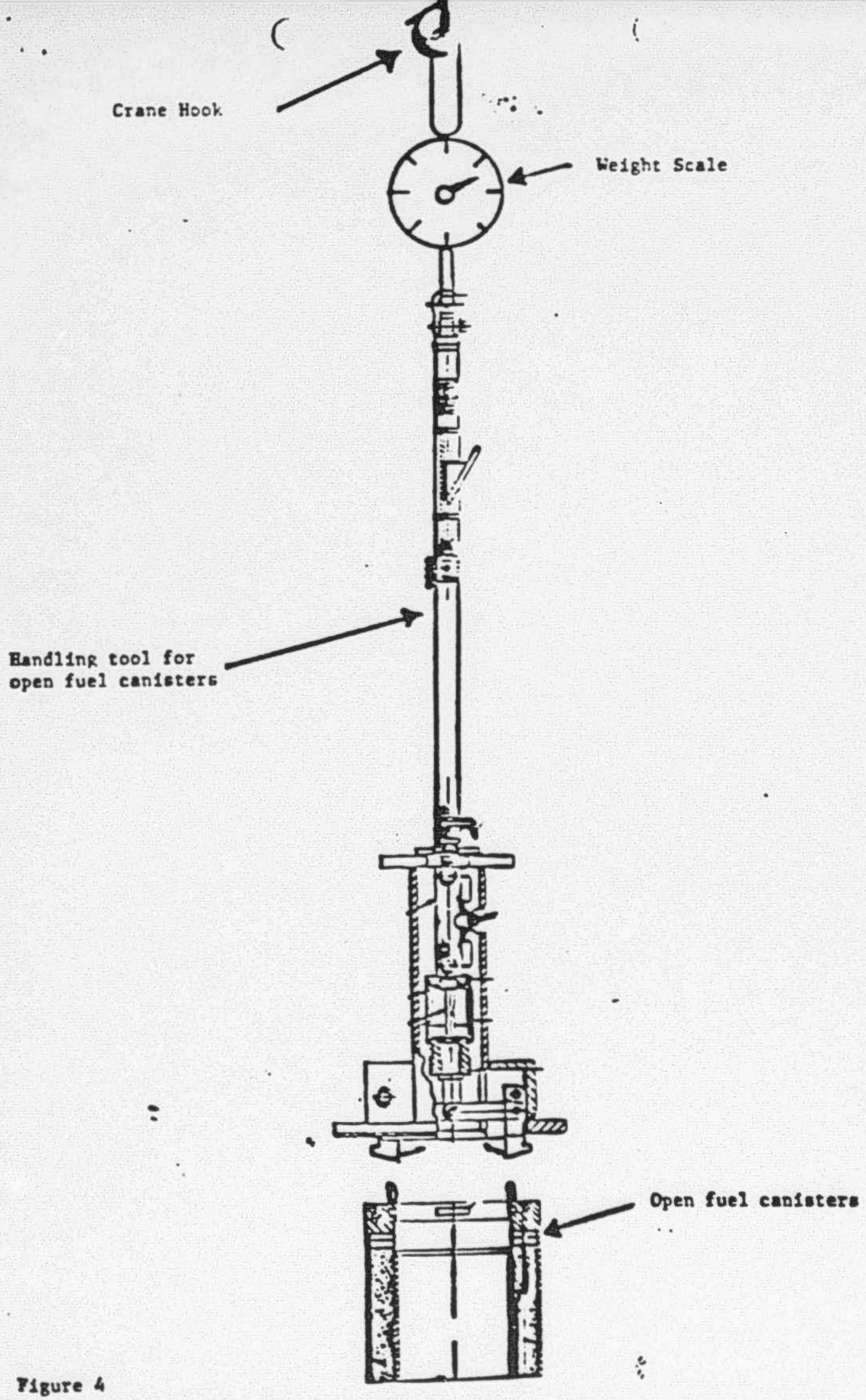
Crane Hook
Weight Scale
Handling tool for
open fuel canisters
Open fuel canisters

Figure 4

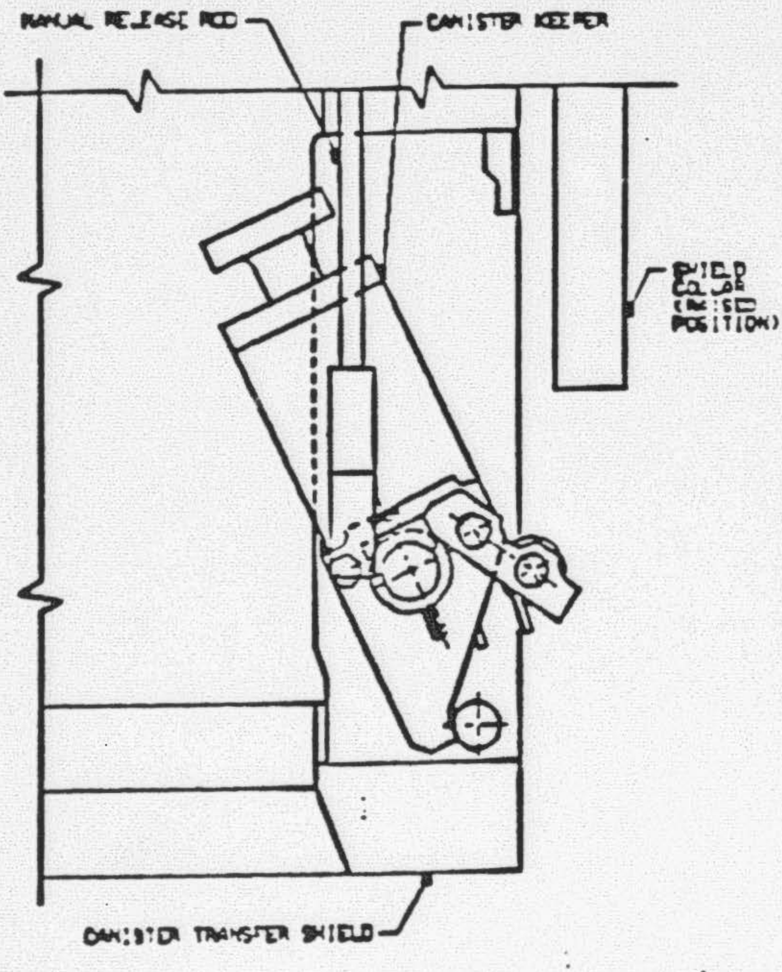

Figure 5: CANISTER KEEPER ENGAGED