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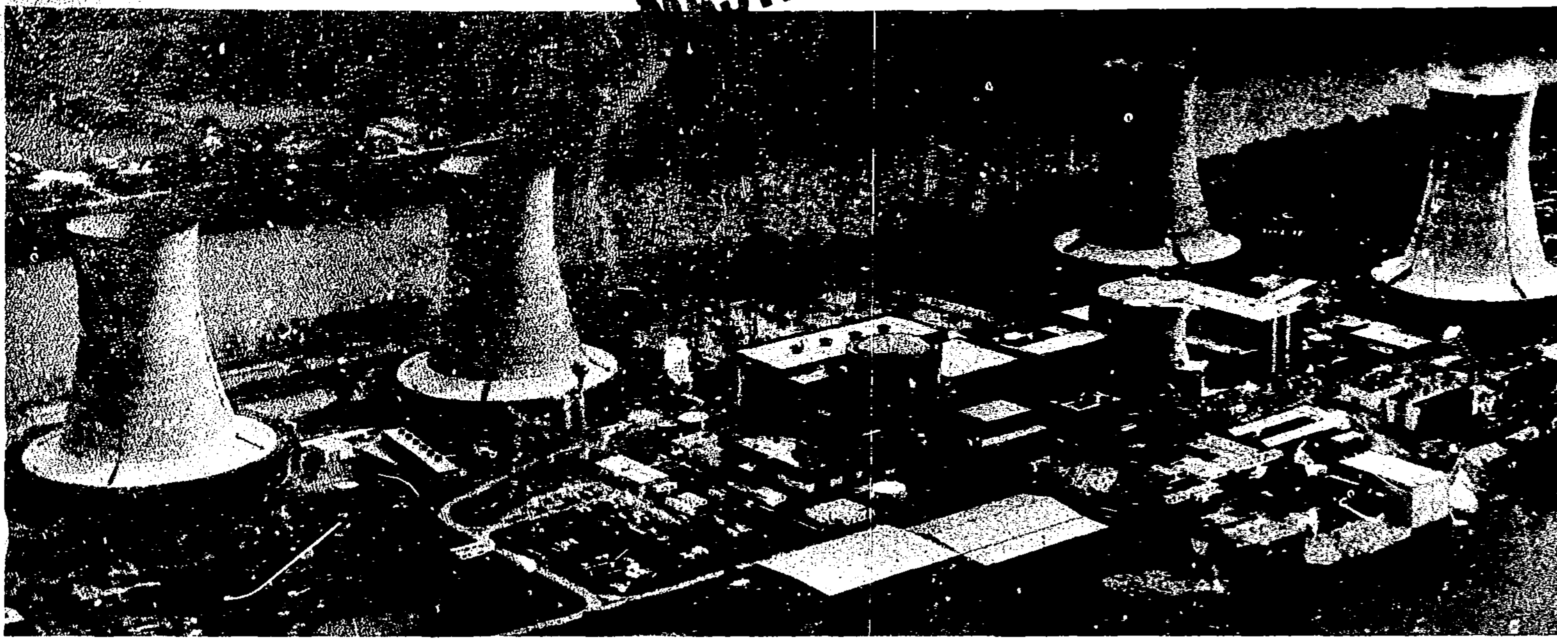

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Canister-Design Considerations For Packaging of TMI Unit 2 Damaged Fuel and Debris

G. A. Townes

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CANISTER-DESIGN CONSIDERATIONS FOR PACKAGING OF TMI UNIT 2 DAMAGED FUEL AND DEBRIS

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FOREWORD

The material in this document was presented by Allied-General Nuclear Services, and subsequently revised in accordance with both discussions that accompanied the presentation and written comments on the earlier version of this document upon which the presentation was based. The presentation was made January 19, 1981 at the Three Mile Island site. Those who attended the presentation and supplied written comments included representatives of EG&G Idaho Inc., General Public Utilities Service Corp./Metropolitan Edison Co., Babcock and Wilcox Co., Bechtel National Inc., Bechtel Northern Corp., and Dominion Engineering.

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ABSTRACT

This report presents the results of a study performed by Allied-General Nuclear Services under contract to EG&G Idaho, Inc., as part of the TMI-2 Information and Examination Program to examine the means of packaging the failed fuel from the TMI-2 reactor core and to provide conceptual canister design. Besides storage and final disposition, a portion of this fuel will be shipped to nuclear facilities to perform detailed physical examinations.

Removal of this fuel from the TMI-2 core is a significant step in the recovery of the facility. The report presents a conceptual fuel canister design. Technical operations are considered to support the design. The TMI fuel when canned will be stored in the spent fuel storage pool. After a period of on-site storage, it is expected that the bulk of the fuel will be shipped off-site for either extended storage or possibly, chemical reprocessing. The final disposition of this fuel, as is common to the expectations for all spent high-level nuclear waste, is geological burial.

Evaluation is made of the technical, economic, and institutional factors associated with alternate approaches to canning of this fuel. A single, multi-application canister is developed into a detailed concept design. Both square and round cross section alternatives are presented. Recommendations are presented concerning other future development tasks whose results could impact the canister detailed design for defueling, canning, on-site storage, and possible off-site shipping of this fuel.

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INTRODUCTION

This report presents an evaluation of technical, institutional, and economic considerations related to the canning of the fuel from the TMI-2 reactor core. This work was performed under contract to EG&G, Idaho by Allied-General Nuclear Services (AGNS). The study takes the form of an evaluation of the various canning options open to the TMI-2 recovery team and the development of a selected canning option into a detailed design concept. Due to the uncertainty of the physical condition of the fuel and its removal from the core, as well as the nationwide uncertainty related to spent fuel disposition, a number of different approaches are covered. The expected time frame for the reactor defueling and fuel canning is 1983 to 1985.

The fuel condition could range from intact to pieces of "debris," or even large "fused" segments of the core. The canning operation must accommodate this wide spectrum of possible conditions. It is expected that the TMI reactor spent fuel pool will be used as a storage area and for other possible handling, testing, and accountability operations preparatory to shipment off-site. The canned fuel will be stored in a spent fuel pool until there is an identified means for disposition of the fuel. A small portion of the fuel will be sent to hot cell areas at national laboratories or commercial facilities to permit a diagnostic evaluation. The remainder of the fuel will eventually be shipped to another location for either interim storage or for final disposition.

The first three sections of the report summarize the results and recommended areas of further work to be performed. The remainder of the study is devoted to the selection and development of the canister concept.

SUMMARY OF RESULTS AND CONCLUSIONS

The major uncertainty in the defueling and canning of the TMI-2 core is, of course, the physical status of the fuel itself. It was necessary to develop several alternative techniques for canning the fuel. In all cases, fuel canning is seen as a necessary condition for the handling of the removed fuel, storing the fuel, and the off-site shipment of the fuel. During handling and storage, packaging is needed for structural integrity, debris collection, and control of pool contamination due to loose debris and soluble fission products. Canning will also be required for shipment to assure containment during transport.

Based on a review of the literature, the fuel condition was assumed as follows:

(1) Three general fuel configurations were assumed. In the first, the fuel is intact but badly weakened and probably bowed in the upper regions of the assemblies. In the second, the fuel is assumed to be debris. Debris may be further divided into two types. The first consists of relatively large pieces which could be handled by mechanical handling devices. The second type consists of smaller pieces which must be vacuumed and filtered. A third fuel configuration assumes that portions of adjacent fuel assemblies may be "welded" to each other in an undefined physical configuration. In this case, techniques will be required to physically separate these larger pieces into sizes which permit canning.

(2) In the central core region, the nonfuel-bearing components such as control rod spiders, axial power shaping rods, etc., may be badly distorted and nonseparable from the remainder of the fuel assembly.

The impact of the radiation environment indicates that standard fuel defueling health physics principles should apply. Full dress anti-contamination clothing should be worn for all work activities carried on in the containment building and the fuel storage pool area at TMI-2. Respiratory protective equipment is indicated for this defueling based on discussions with GPU and EG&G personnel. However, the impact of the radiological environment appears to be a secondary one on canister design. Routine principles of minimizing fuel handling, standardizing and simplifying operations, and minimizing equipment transfers across radiation boundaries are considered in the canister alternate selection process and the selected alternate design development.

Of course, the defueling and associated canning of the fuel will be done in a manner that precludes a recriticality configuration of the TMI-2 core. Of direct impact on the canning system scoping and selection process, is whether the sectioning of "welded" fuel assemblies must be restricted in any manner that could influence the canister design. The envisioned sectioning method or apparatus was not developed but its results would be configurations approximating intact fuel assembly envelops or portions thereof. AGNS analysis of a reference case collaborated with results in the literature and established that such restrictions did not exist.

Generically, three canister alternates were considered that included:

(1) Single Multi-Application Canister
(2) Multiple Single Purpose Canisters
(3) Inner Shroud/Outer Shipping Canisters.

A basic selection criterion was then applied to the alternatives. So long as a thorough definition of the problem based on the available data does not preclude a simple, single solution, then this approach should be selected, particularly if it bounds a concensus of viewpoints of the unknowns.

Such is the case with the single, multi-application canister. The spectrum of core conditions could be accommodated with this canister concept and operational conditions imposed by the radiological environment do not preclude its use. Further, since "sectioning" is not restricted by criticality considerations this single, multi-application canister is compatible with envisioned "sectioning" procedures.

The single, multi-application canister also indicates lower costs both from an inventory as well as a manufacturing set-up point of view. For these reasons, the single, multi-application canister was selected for concept design development. However, to permit a selection process (beyond the scope of this effort), two alternatives for the single, multi-application canister are presented. One is a canister with a critically safe, square cross section and the other is a canister with a round cross section that requires poison for criticality safety in water-reflected environments. The round canister cross section maximizes the loading cross sectional area within the constraint of a fixed diagonal or, in this case, a diameter. This, in turn, is expected to minimize the potential for the canister wall to interfer with the damaged fuel envelop during canister loading.

Development of the selected canning system alternate resulted in the square and round concept design alternates. They accommodate the full spectrum of assumed core conditions:

(1) Intact (plus non-fuel bearing components)
(2) Debris (chunks to fines)
(3) Sectioned "fused" core.

They also accommodate a variety of projected fuel handling tools and loading modes.

These single, multi-application canister alternates are 177 3/4 inches long. The square alternate has a 9.16-inch inside dimension; the round alternate an approximate 12.25 inches inside diameter. These dimensions provide for thermal expansion. The square alternate has a 6000-pound design capacity; the round a 8500-pound design capacity. The square alternate has 5/8 inch side clearance to the unirradiated fuel width and the round at the fuel corners would have an approximate 1/4 inch side clearance. These dimensions are also compatible with three potential legal weight truck (LWT) casks; the NAC-1, the NLI 1/2, and the Fort St. Vrain. The side wall thicknesses were selected on the basis of various structural criteria and were shown acceptable to postulated accident conditions. Internal pressurization of the square canister at less than 25 psi ΔP and the round canister at less than 200 psi ΔP should not produce stresses in excess of yield on the 304L stainless steel material. This material appears adequate for all corrosion potentials. The unit cost for an order of 250 units is estimated to be $2200 each for the square alternate and at least $3000 for the round alternate. Dimensional constraints that were applied are identified in Table 1.

Using "worst case" water-reflected, criticality assumptions (i.e., using, for example, an unborated water reflector), the K_{eff} of the isolated square canister was shown to not exceed 0.955. Similarly, the isolated poisoned round canister was shown to not exceed 0.946.

In an array, a square or round canister will be essentially "isolated" from its neighbor provided that 8 to 10 inches of water is placed between units. The array K_{eff} would then be no higher than that of an individual unit. The calculated K_{eff} of an infinite array of poisoned round canisters spaced closer with two inches of water separating adjacent units was shown to not exceed 0.970. Additional rack poison is not expected to improve round canister array spacing but could be effectively used in square canister array spacing optimizing storage space against rack poison costs.

The closure of the square canister alternate is achieved by pressing the crane-grappled cap into the top of the canister. In the center of each side of the cap, keys are provided. The sides of the canister are grooved to accept these keys. When the cover is pressed into the top of the canister, the sides of both the canister and cap yield enough to permit the keys to enter and then expand into the grooves on the sides of the canister. Preliminary analysis indicates this closure fastening method could be implemented using allowables that provide a factor of safety of three on yield when lifting the canister by the cap. "O" ring or fusible inserts could be provided for mechanical sealing until cap seal welding could be completed.

The round canister cap is screwed and provides an equal safety factor.

The pressure relief valves used in these canisters are mounted to preclude handling damage and to prevent inleakage of water if the canisters are purged as would be required for transport in a dry shipping cask or for dry storage. To prevent the escape of gas pressure generated inside the canister, the closure end of the valves must be plugged. The plug boss on the cap is designed to accept a device used for sampling the gas pressure and contents of

TABLE 1. DIMENSIONAL CONSTRAINTS

		Constraint	
SQUARE:	Actual	Origin	Dimension
Outside Diagonal	13.250"	LWT NLI 1/2	13.375
Wall Thickness	0.135"	Horiz. to Vert. Bending Stress <10K psi	
Outside Width	9.555"	(see outside diagonal plus radii corner)	
Inside Width	9.160"	Fuel Cross Section	8.576
Length (Inside Cavity)	171"	Fuel + CRA	170.75
Length (Max. Envelop Outside)	175"	LWT NLI 1/2	175.22
Length (Overall)	177.75"	LWT NAC	178.00
ROUND:			
Diameter Outside	12.625"	LWT NLI 1/2	12.875 (minus 1/2"Ø drain)
Wall Thickness(W/O Poison Sleeve)	0.160"	Horiz. to Vert. Bending Stress <10K psi	
Diameter Inside	12.141"	Fuel Chamfered Diag.	11.953
Length (Inside Cavity)	171"	Fuel + CRA	170.75
Length (Max. Envelop Outside)	175"	LWT NLI 1/2	175.22
Length (Overall)	177.75"	LWT NAC	178.00

the canisters without loss, if this information is desired. A porous stainless steel filter plate is welded over the inlet of this boss so that if a continuous vent storage method is selected for the fuel, the plug could be removed and the filter vent would satisfy this requirement. A connection could be made to collect any effluent from the canisters by connecting to the threaded hole after the pipe plug has been removed.

RECOMMENDATIONS

As is noted in the discussions of Canister Design section following, various additional information is needed prior to finalizing the canister design. The following identifies these requirements. Once the Canister Prototype below is finalized, an orderly defueling program would proceed with the other items below.

Canister Prototype

Detail design and fabrication of a canister prototype should be implemented following selection of a design concept. This will verify manufacturing methods, identify design revisions that would facilitate volume manufacture, refine cost estimates, and refine mechanism development such as mechanical fitup of cap and canister. Detail design is estimated at three months and prototype fabrication at six months if poison is required, otherwise four months. A production run is estimated at ten months.

Core Mockup

Once a prototype canister is available, a core or partial core mockup (e.g., one quarter) should be designed and constructed at a cold facility to develop canister remote operational features (e.g., canister capping, handling, and sealing) as well as canister support tooling. Then, using dummy fuel constructed to approximate projected core damage, defueling techniques could be developed. These would include intact fuel lifting, large debris handling, hydraulic suction, and "welded" core sectioning equipment. Once the necessary tooling development is being finalized, defueling crew training could commence.

Storage Rack Design for Canned Fuel and Debris

The outside dimensions of the developed canister design necessitates new (i.e., canned fuel) storage racks. Also, the design canister loadings of debris from multiple assemblies in one canister alters considerably the normal "design basis fuel" for the rack criticality and seismic considerations. These facts suggest an early start for the design and development of a "state-of-the-art" storage rack design for the fuel canisters. Preliminary evaluations should commence as soon as possible since rack licensing could take three years or more to implement.

Canister Capping Robot Welder Development

No commercial device exists to weld the cap to the canister underwater as discussed later without modification. A development program should be implemented to adapt existing devices for this special application or other cap sealing methods should be explored that do provide essentially permanent hermetical sealing.

CANISTER DESIGN

Radiation and Core Condition Impact Assessment

In the previous AGNS effort on dispositioning the TMI-2 core,[1] the major uncertainty in the defueling and canning of the TMI-2 core was identified, of course, as the physical status of the fuel itself. However, the conclusion was made that canning of fuel would be required as a necessary condition for handling the fuel, storing the fuel, and the off-site shipment of the fuel. In the former cases, packaging is needed for structural integrity, debris collection, and control of pool contamination due to loose, nonsoluble debris, and soluble fission products. Canning will also be required for shipment to assure containment during transport.

The canning conclusion of AGNS (see Reference 1) was based on the following core condition assumptions:

"(1) Three general fuel configurations were assumed. In the first, the fuel is intact but badly weakened and probably bowed in the upper regions of the assemblies. In the second, the fuel is assumed to be debris. Debris may be further divided into two types. The first consists of relatively large pieces which can be handled by mechanical handling devices. The second type consists of smaller pieces which must be vacuumed and filtered. A third fuel configuration assumes that portions of adjacent fuel assemblies may be "welded" to each other in an undefined physical configuration. In this case, techniques will be required to physically separate these larger pieces into sizes which can be canned.

(2) In the central core region, the nonfuel-bearing components such as control rod spiders, axial power shaping rods, etc., may be badly distorted and nonseparable from the remainder of the fuel assembly."

Core Condition Impact

In the current effort, more recent literature was referenced to determine whether the previous core condition conclusions should be altered. NUREG-0683[2] essentially bracketed the core condition assumptions of the previous effort using its "Best-Case" and "Worst-Case Conditions" format.

Likewise, the GEND-007 report[3] substantiated these conclusions of core condition. The GEND-007 report was a comprehensive effort to review and summarize the core damage assessments which have been made in the literature to identify the minimum and maximum bounds of damage, and to establish a "reference" description for the current status of the core. The conclusions of AGNS (Reference 1) essentially paralleled the minimum and maximum bounds of damage detailed by the GEND-007 report.

The following is quoted from the summary of the GEND-007 report.

"Factors of primary interest during reactor disassembly and removal of the core are the condition of the upper plenum, the amount of cladding oxidized, the presence of once molten materials such as liquidified fuel and control rods, and the condition of the instrument and guide tubes. Some components of the upper plenum structure may have melted or fused together during the course of the accident necessitating the development of tooling and procedures for this contingency. It is evident that a bed of fragmented fuel and cladding has formed, perhaps extending to the core periphery. A few of the upper plenum components may rest on top of the debris. The amount of cladding oxidized, approximately 50%, is indicative of the fraction of the core which is brittle or fragmented. The presence of liquified fuel, or any once molten material, is enough to ensure that some areas of the debris will be fused together and that separation techniques and tools must be designed accordingly. The total weight of potential debris and embrittled cladding is 64,000 to 83,000 kg (140,000 to 184,000 lb)."

Other publications, such as NSAC/EPRI 80-12 referenced elsewhere[4] were independently investigated also.

In summary, retaining the core condition conclusions of AGNS (Reference 1) as quoted earlier appears compatible with the conclusion of the references reviewed. Therefore, these conclusions require a canning system that accommodates (1) intact fuel, (2) debris, and (3) sectioned fused debris with the assumption that intact fuel may have nonseparable nonfuel-bearing components such as control rod assemblies, axial power shaping rods, etc.

Radiation Impact

It is projected that when proposed TMI-2 defueling/canning operations occur, ambient radiological conditions will not significantly impede operations. The projection is based on radiation survey data collected during recent containment building entries and the reported spectrum of radionuclides present. During defueling/canning operations, the fuel handling canal will be filled with water the same as it would be during normal refueling.

The impact of the radiation environment indicates that standard fuel defueling health physics principles should apply. Full dress anticontamination clothing should be worn for all work activities carried on in the containment building and the fuel storage pool area at TMI-2. Respiratory protective equipment is indicated for this defueling based on discussions with GPU and EG&G personnel. However, the impact of the radiological environment appears to be a secondary one on canister design. Routine principles of minimizing fuel handling, standardizing and simplifying operations, and minimizing equipment transfers across radiation boundaries were considered in the canister alternate selection process and the selected alternate design development.

Criticality Safety During Defueling

Of course, the defueling and associated canning of the fuel will be done in a manner that precludes recriticality of the TMI-2 core. Of direct impact on the canning system scoping and selection process, is whether the sectioning the "welded" fuel assemblies must be restricted in any manner that could influence canister design. The envisioned sectioning method or apparatus was not developed, but its results may be configurations approximating intact fuel assembly envelops or portions thereof. Analysis established that such restrictions did not exist.

Criticality analysis of the TMI-2 core has been made by the NRC staff,[5,6,7,8,9] Babcock and Wilcox,[10] Brookhaven National Laboratory,[11] GPU,[12,13] and ORNL.[14] These analyses assumed various disruptive states of the TMI-2 core. Collectively, the analyses conclude that with a boron level of 3500 ppm, the core will remain safely subcritical for any physically reasonable rearrangement of the fuel, even assuming the total absence of control rods and burnable poisons. Subsequent analysis by AGNS supported this conclusion. The AGNS analysis utilized the NITANL/XSDRNPM/KENO-IV computer programs with the XSDRN 123 energy group neutron cross section set.

Canister Alternates

Generically, three canister alternates were considered to include:

(1) Single Multi-Application Canister
(2) Multiple Single Purpose Canisters
(3) Inner Shroud/Outer Shipping Canisters.

The alternative selected must accommodate the core condition conclusions noted in the Radiation and Core Condition Impact Assessment section above (i.e., basically intact fuel assemblies with or without nonfuel-bearing components, broken sections of fuel assemblies, sectioned fused debris to approximate fuel assemblies or portions thereof, and shapeless debris such as pellets or portions thereof). The alternative should also accommodate a spectrum of loading tooling and orienations. The tooling could include normal intact fuel grappling tools, special bottom lifting tools for intact fuel that engage internally or externally to the fuel envelop, unarticulated hooks and articulated grapple equipment for debris, and hydraulic suction apparatus. Loading orientations could include gravity vertical or top loading, assisted horizontal loading and vertical loading into an inverted canister positioned, for example, just above an assumed intact core periphery fuel assembly.

The following discussion is an early effort intended to show the interaction of the various alternatives with potential tooling. The intent is to show that the development of the alternative as presented is not contrary to a reasonable spectrum of support tooling. It also provides a catalyst to develop dialogue on can interface tooling. It is not intended to specify or limit alternative tooling concepts under development or study by other groups which have been advanced beyond the conceptual level of the following discussions in more recent efforts.

Single Multi-Application Canister

An end capping canister, that exceeded the design-fuel-plus-nonfuel-bearing components envelop by an acceptable tolerance, provided a single candidate canister that could accommodate the anticipated spectrum of fuel conditions, loading tooling, and loading orientations discussed above. Various examples of selected cases in the above spectra are discussed below.

It should be noted that review meeting conclusions discussed in the "Foreword" established that the square canister shown in the various illustrations should be considered an alternative of the single multi-application canister. An additional alternative of the single multi-application canister is one of circular cross section which requires some form of integral poison. A conceptual design of this alternative is developed in the Single Multi-Application Canister Design Development section and should be considered as an alternative for the square cross section wherever the square cross section is shown or discussed in the following in regards to the single multi-application canister design.

The single multi-application canister is shown in Figure 1 oriented for vertical gravity loading of intact fuel grappled normally, broken portions of fuel assemblies, or sectioned debris that could be loaded with a variety of lifting tooling. The debris funnel of Figure 2 would be optional but should simplify the remote operation.

This canister could also be loaded horizontally with intact fuel as discussed below. This canister-loading system requires that a minimum of two fuel assemblies be removed either by conventional or other means before the special bottom lifting, fuel removal tool can function properly.

The special bottom removal tool of Figure 3 recognizes that the initial lift to free the fuel from its socket in the core plate requires more lift strength of the already questionable fuel assembly structure than just the free weight of the element in the coolant water. This device lifts the fuel element from the core and assist in placing it in a storage canister with only compressive gravity stress being applied to the fuel element structure. This system minimizes the risk that the fuel element or some parts of it might come free while the element is being lifted, transported, and canned.

Figure 4 shows the detail of the engagement of the lift and handling fixture to the fuel. A box beam is employed as a strong-back to apply the lift force to the bottom of the fuel element. A double-pronged engagement piece is welded to the bottom of the box beam and is designed to slide under the fuel element. The raised edges are fastened to the fuel engagement side of the box beam to provide a shallow trough in which the fuel element is restrained. A lifting

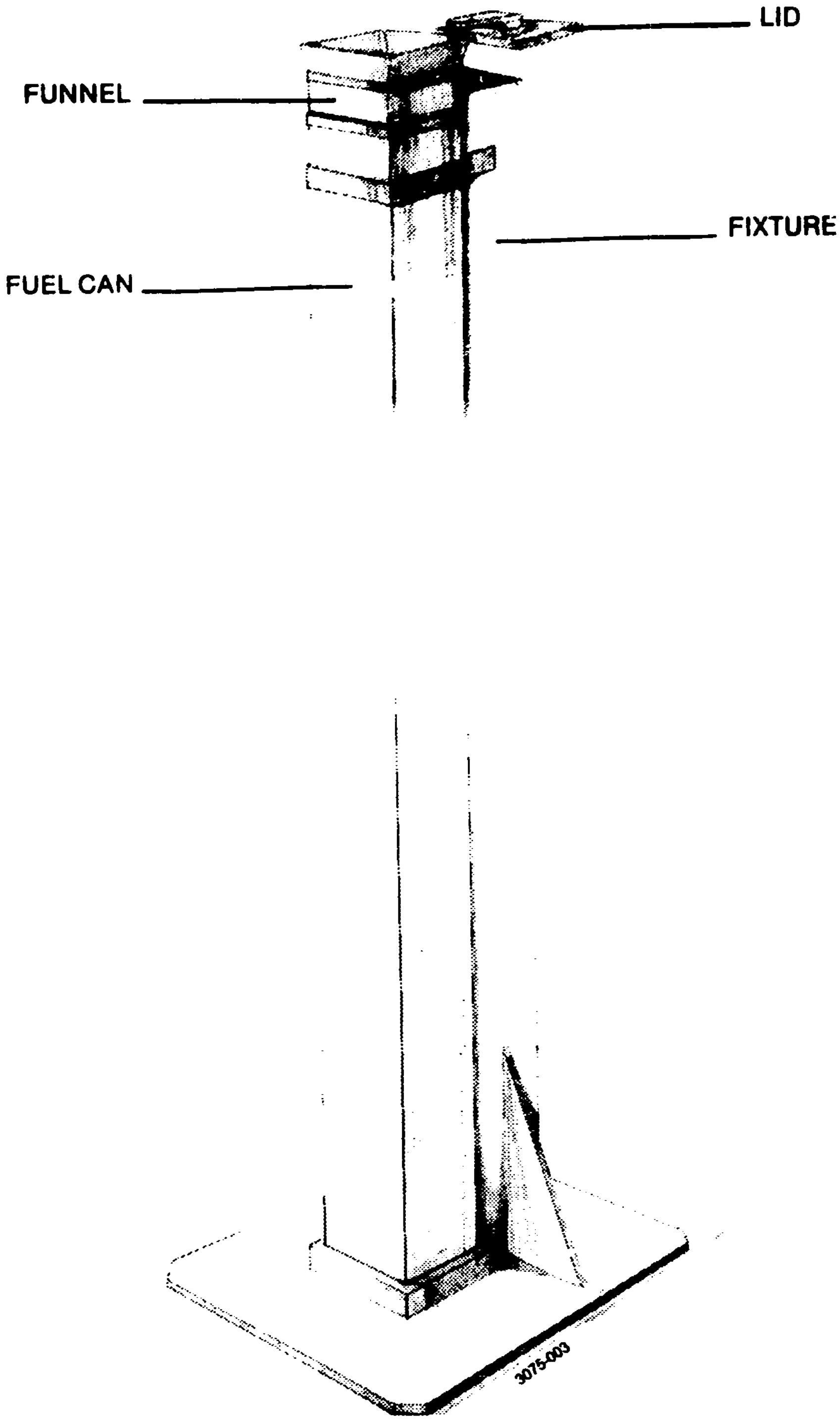

Figure 1. Single multi-application canister.

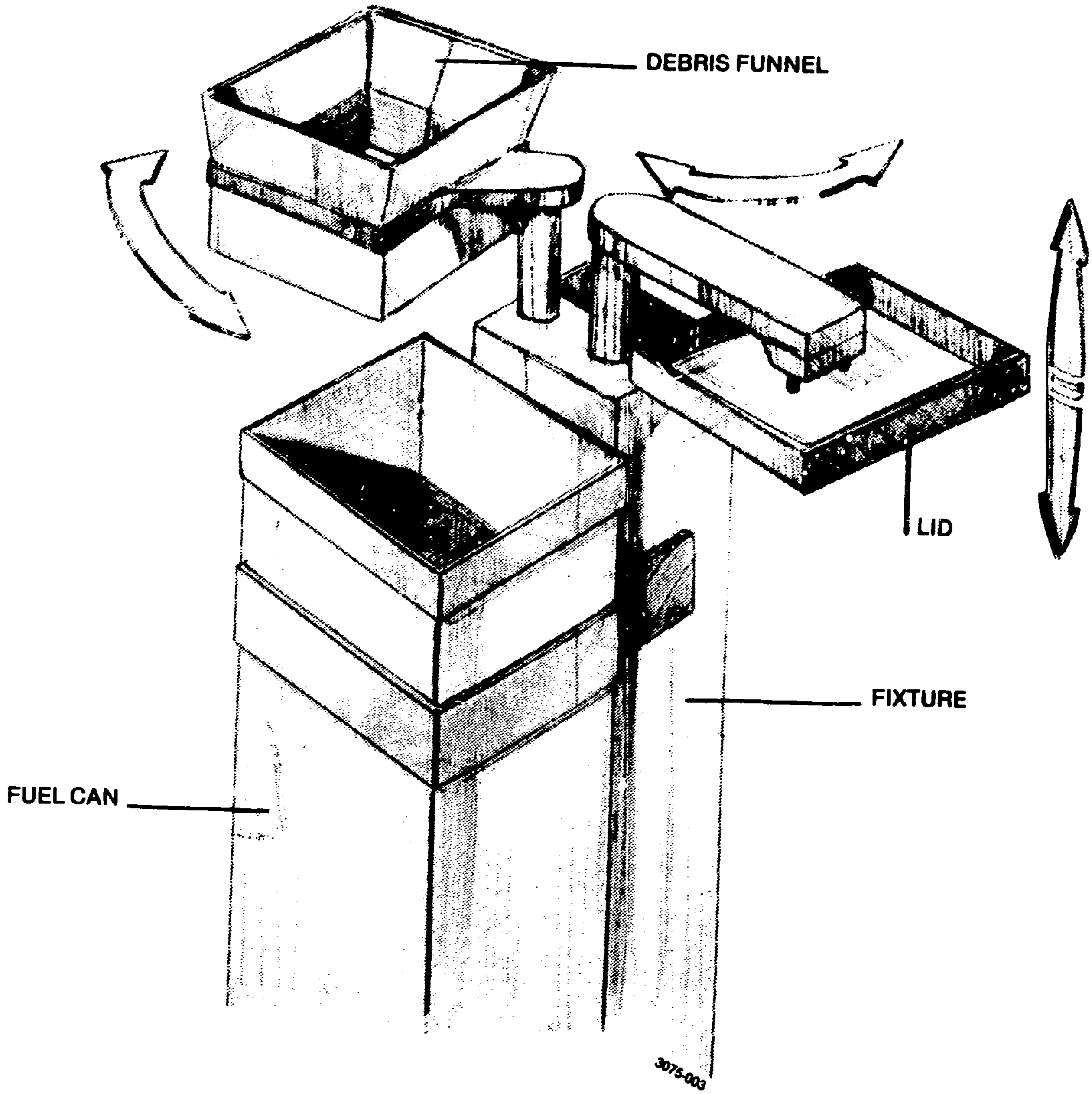

Figure 2. Debris funnel for single multi-application canister.

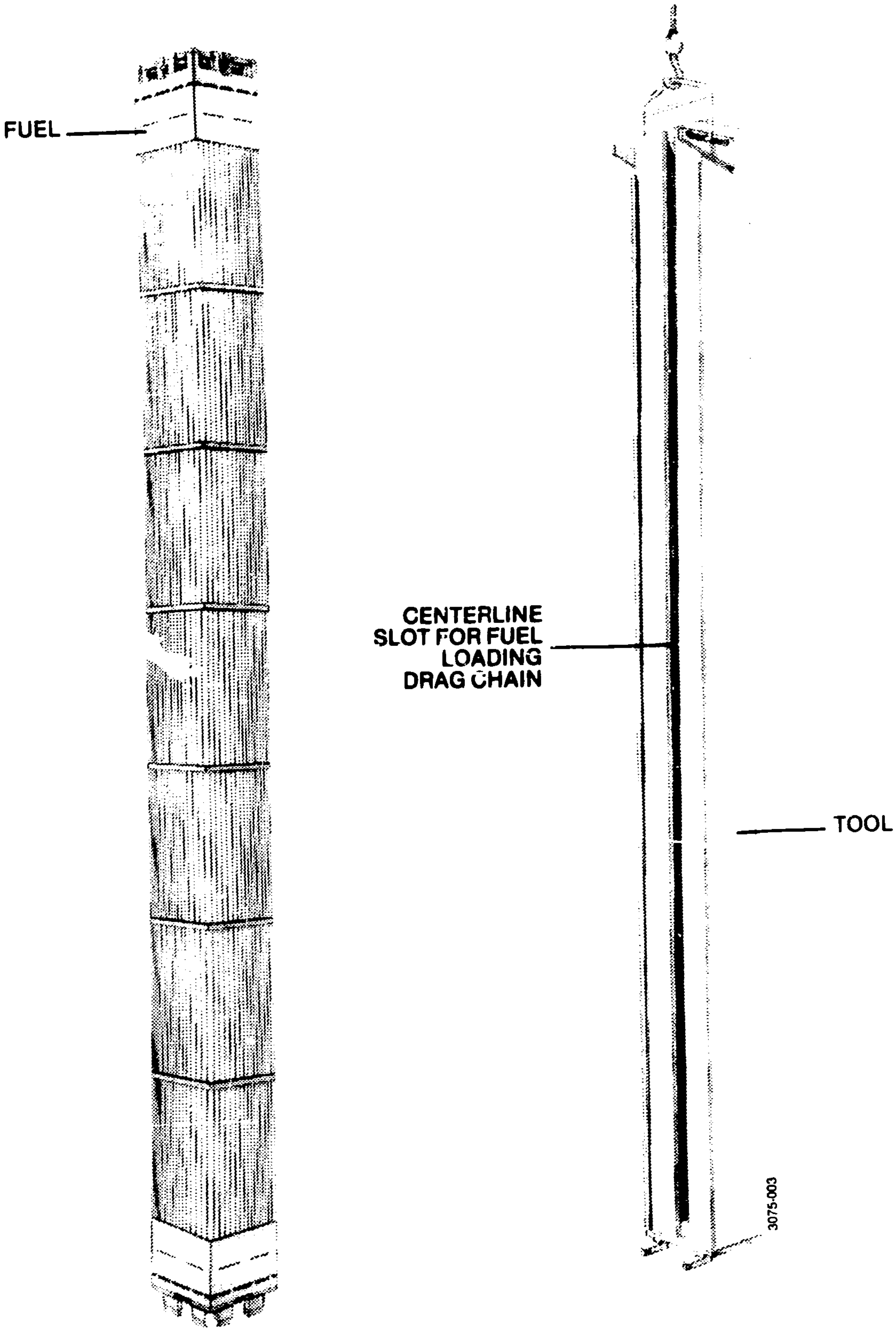

Figure 3. Special bottom removal tool.

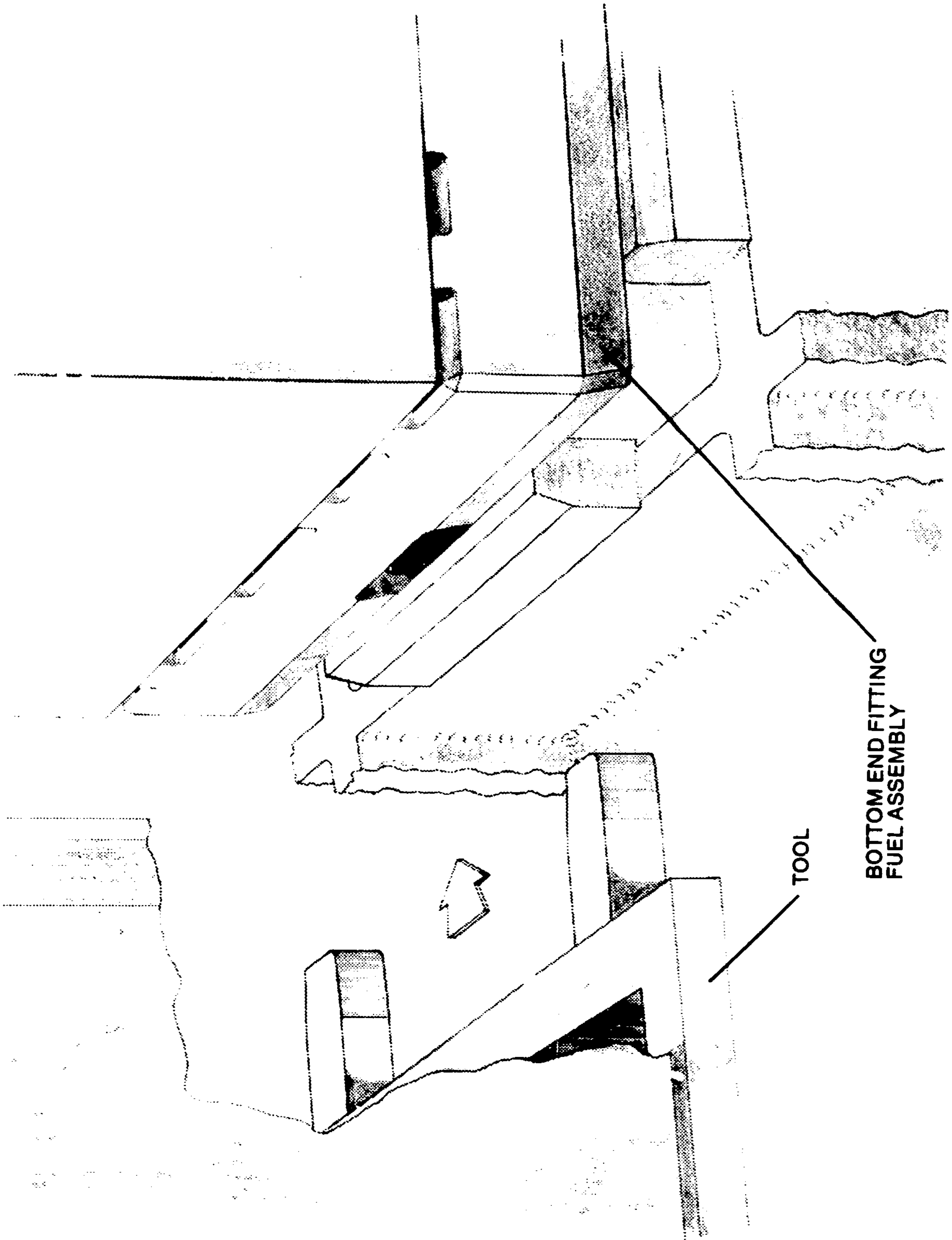

Figure 4. Special bottom lift tool engaging fuel.

bail is provided at the top of the assembly which will center over the center of gravity of the lifting tool when it is empty and then slide to the "fuel-plus-tool" center of gravity after the fuel element is engaged. A latch would be provided to insure that the fuel element cannot escape from the fixture during the handling operation.

In operation, this device would be attached to a crane and lowered to the core support plate in vacant spaces in the reactor core where at least two elements have been removed. Even if these spaces cannot be cleaned initially by full assembly removal, contingency methods will eventually establish this condition. Various horizontal pusher type auxiliary tools could be needed to seat the engagement projections under the fuel element. The projections could be shaped so that as they are pushed into position they wedge the fuel element loose from its socket. After placement, the lift bail is moved to the "fuel-plus-tool" center of gravity. The lift would be made and the fuel deposited in the pivoting fixture and "tilted" to a horizontal position (see Figure 5).

The single multi-application canister is placed on the loader in front of the lifting device that is now horizontal. A drag chain device in the box beam of the lifting tool (not shown except for its centerline slot in Figure 3) is activated to push the fuel element into this canister. Contingency methods would have to be implemented in the event of a jam during loading. The loaded canister is retracted from the lift device by the cylinder and a canister cover is positioned by the cover holder (not shown). The canister, with the fuel element inside, is moved toward the lift device so as to press the canister cover in place on the canister. The canister is once again retracted from the lift device and is ready to be transported through the transfer tubes to the storage area.

In whichever pool is selected, the canister could be lifted vertically by the bail on its cover. The cover could be seal welded underwater by a robot welder using a bonnet from which the water has been purged. (Alternatives to the bonnet such as fixed, dry chambers might also be considered.) Water could be removed from the canister by applying air pressure through threaded connections on the canister cover. Water inside the canister could be collected by attaching a line to the threaded connection on the bottom of the canister. After the

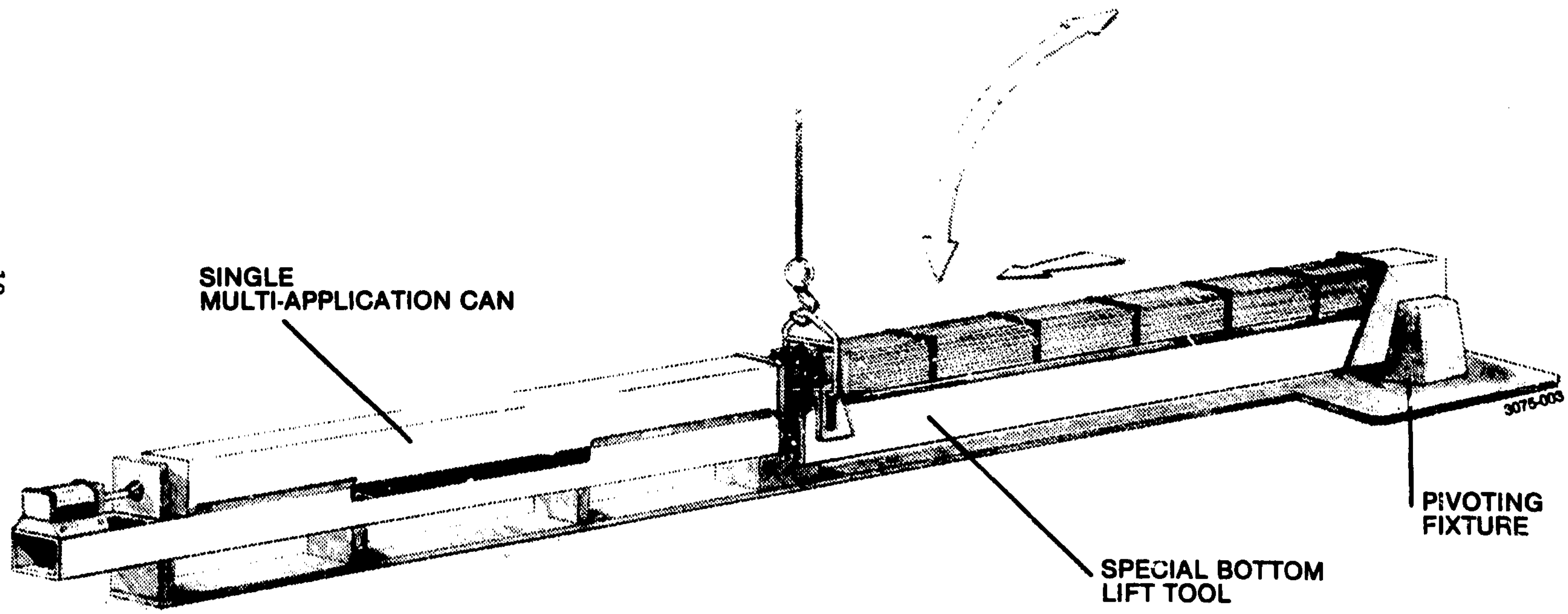

Figure 5. Pivoting fixture.

water is purged from the canister, the lines can be removed from the canister and the check valves would prevent water from entering the canister. If a permanent seal was desired, the outlet to the drain lines could be plugged with luted pipe plugs.

An additional tooling and loading orientation alternate for canning intact fuel in the single multi-application canister is shown in Figure 6. This approach minimizes the handling of the fuel to the maximum extent possible. The tooling positions the canister in the core barrel just above the fuel to be removed. Similar requirements exist to provide an empty fuel position or two for tool positioning and to engage the tool to the fuel as discussed in the bottom lifting/horizontal loading alternate discussed above.

The tooling is a telescoping device with one portion holding the canister and the other engaging the fuel. A fuel loading force provided by the crane exerts a downward force on the canister and an upward force on the fuel due to the tool cable reeving. The tool would be counterweighted so that the tool weight would exceed anticipated frictional forces during loading ensuring the full seating of the fuel in the canister prior to lifting of the loaded canister to the capping station. Note that initial fuel movement immediately advances the fuel within the known structural integrity of the canister minimizing potential increases in the existing damaged condition of the fuel that could result from unnecessary handling.

The single multi-application canister also handles that spectrum of shapeless debris such as pellets or portions thereof. Loading methods here could include hydraulic suction and inertial, mechanical, and gravity separation of the particulate from the liquid. The canister would serve as a collection bag. The collection bag function conventionally would be provided as a gravity exit from a positive displacement pump motivated vacuum collection system as shown in Figure 7. A cyclone separator as shown would increase the separation efficiency.

Other collection bag approaches for the canister could include more direct loading by using less efficient gravity separation. They could eliminate some equipment by utilizing inserts for the canister that provide staged settling

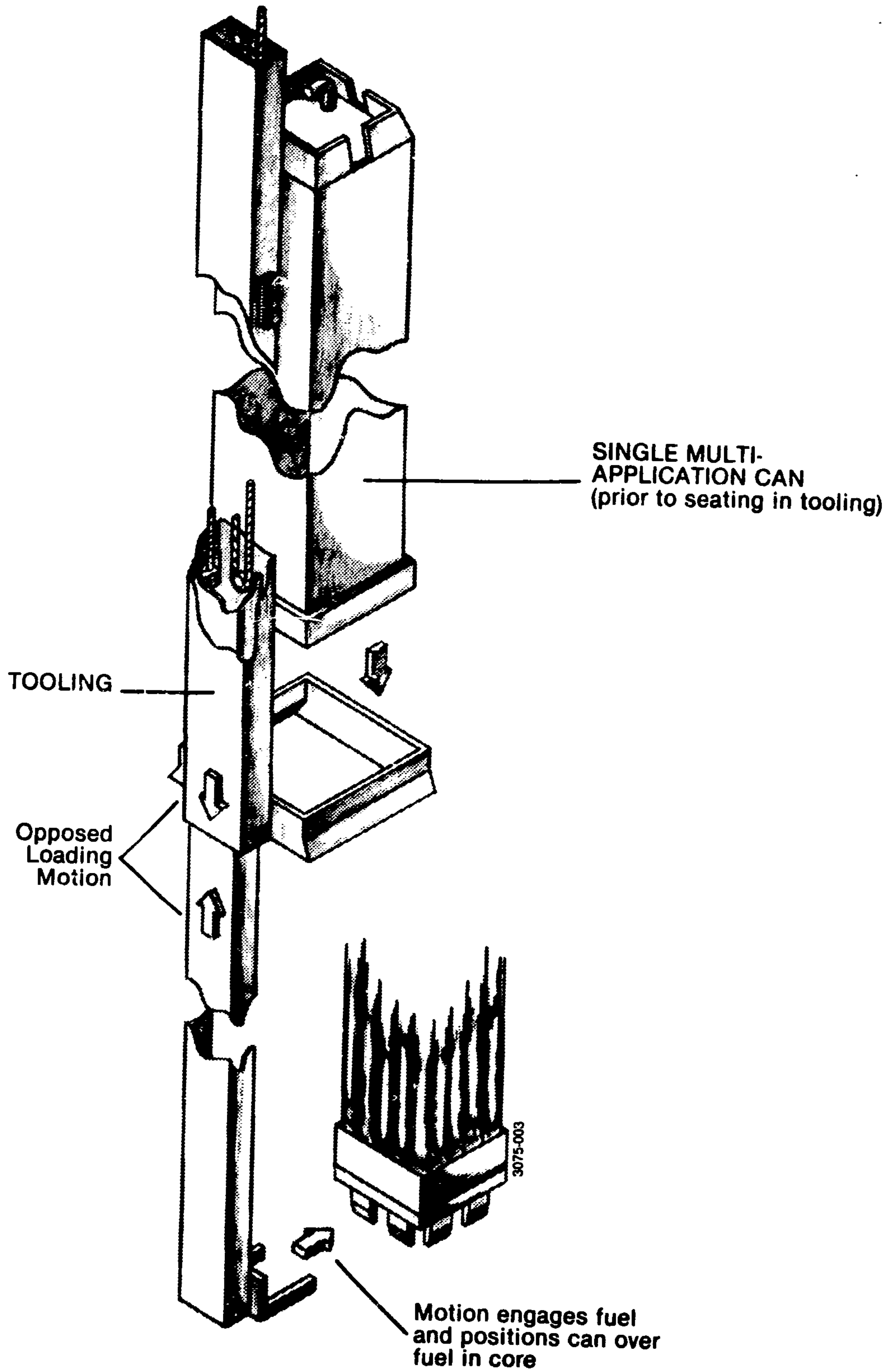

Figure 6. Over-core loading tooling.

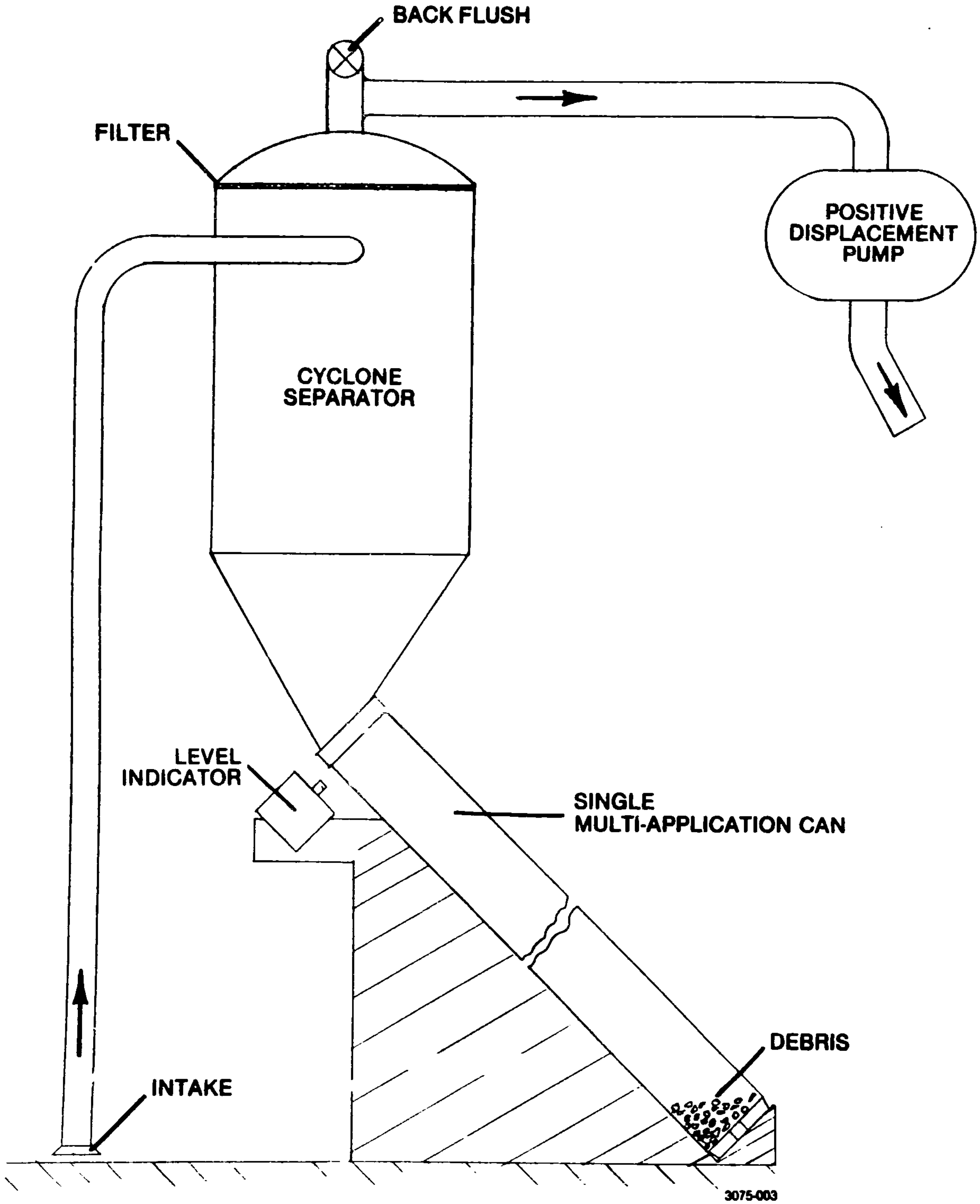

Figure 7. Vacuum system with single multi-application canister as a vacuum bag.

chambers as shown in Figure 8. The insert shown would be placed within the canister. The vacuum nozzle must be connected at one end of the canister and the positive displacement pump connected at the other end. This approach would require modification of the canister top and bottom to mate with the connectors. The canister orientation would be vertical but similar horizontal inserts can be provided.

The above discussion indicates that for a spectrum of fuel conditions and loading tooling and orientations, an end-capping canister that exceeds the design-fuel-plus-nonfuel-bearing-component envelop by some tolerance appears to be an acceptable candidate for a single, multi- application canister for TMI-2 fuel.

Multi-Single Purpose Canisters

Various examples of single-purpose canisters could be conceptualized that provide application for specific potentials. This approach is useful where a limited number of well defined categories of fuel condition exists, specific existing tooling and loading methods must be utilized, or specific dispositions are well defined.

An example would be a canister of dimensions exceeding those of a routine fuel canister designed for a limited quantity of fused debris, transportable by a specific cask, to be disposed of in a specified manner.

Pending future developments in core condition data, disposal option, etc., this approach does not appear prudent so long as a reasonable potential spectrum of core, loading, and disposition options are accommodated by a single multi-application canister. Therefore, development was not directed further in this area.

Inner Shroud/Outer Shipping Canisters

In the previous AGNS effort (see Reference 1), consideration was directed at providing a shroud as soon as possible for the fuel during defueling. The shroud was intended to only enhance the questionable structural integrity of

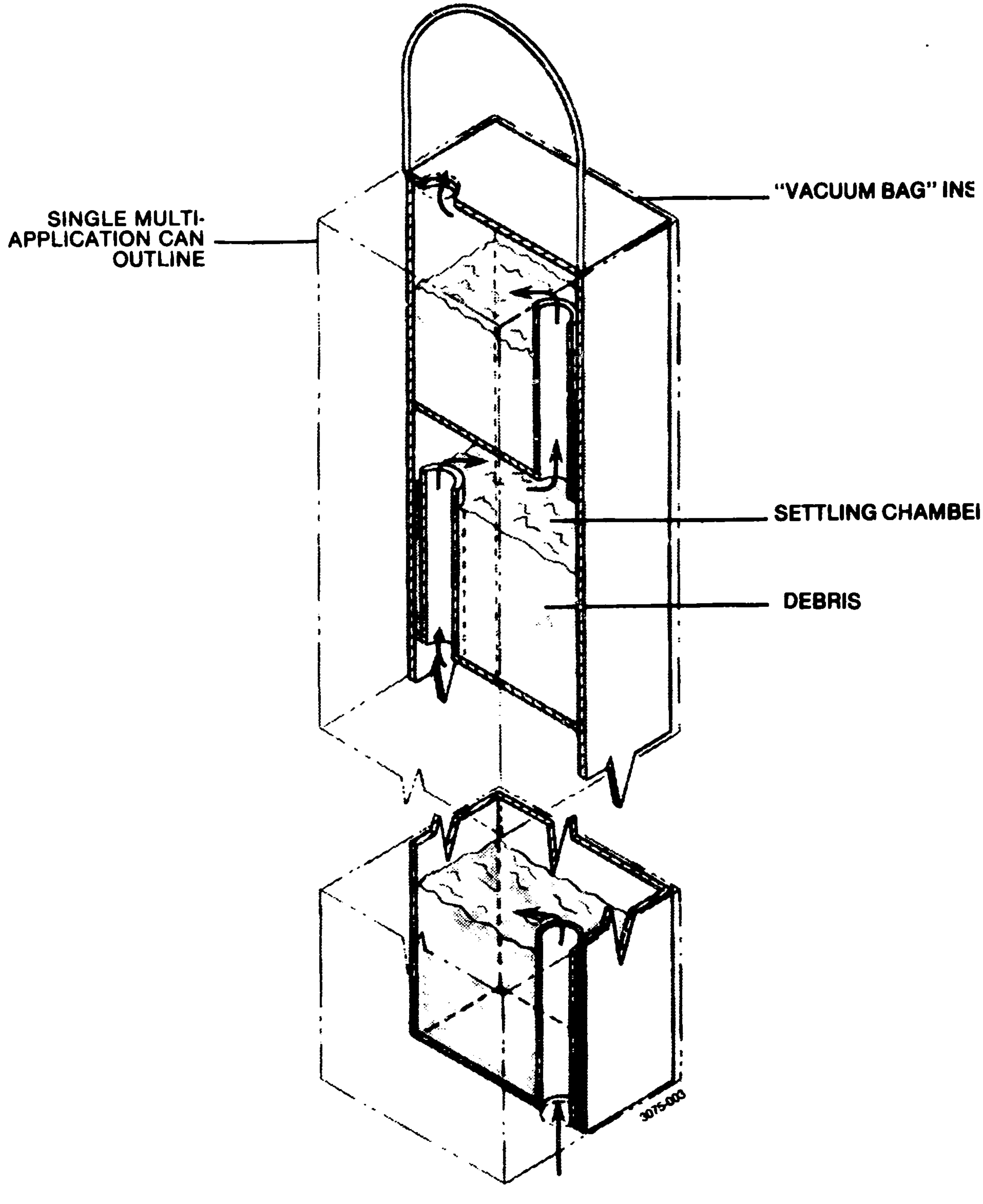

Figure 8. Vacuum bag insert.

the fuel. Since the shroud is not intended to confine gaseous or particulate fuel contamination, extended storage and transport of the fuel would require an additional structure or outer shipping canister to be placed over the inner shroud. Examples of this basic concept are discussed below. First, the shroud approach for intact fuel and its modification for debris are discussed. Then the outer canister concept is detailed.

A handling shroud concept shown in Figure 9 would accommodate top grappled fuel. The shroud would be "bottom loaded." Once the fuel was lifted completely, the pads (see Sections AA and BB of Figure 9) would drop down and could support the fuel if needed. The need would arise if axial failures of the fuel occurred during further handling.

The intended handling sequence is shown in Figure 10. The grapple head would be placed within the shroud. The grapple cap would be secured and the grapple would move the shroud into its holder for loading. This would permit the grapple to lower and to engage the fuel. Meanwhile, the shroud is being supported by the holder. Sequence 1 to 3 shows shroud loading.

It might be desirable to let a portion of the grapple remain with the fuel. This would require a disconnect designed into the grapple. But, at this stage of the study, a reusable grapple was assumed. The next step would require setdown of the shroud. Sequence 4 of Figure 10 shows the shroud after setdown in the fuel transfer carriage basket. In Sequence 4, the grapple cap is shown being removed, thus freeing the grapple which would likewise be removed. The grapple cap detail can be seen in Figure 9 showing the release pin and removal hinge. The handling cap is then installed (Sequence 5). Finally, in Figure 10, Sequence 6, the fuel shroud is shown after being loaded into a canister and ready for shipment.

A side access shroud which would accommodate the fuel lifting method for badly distorted fuel is shown in Figure 11. The fuel would be side loaded into one of the 90° half-sections. A holding frame would incline this section to ensure that the fuel did not topple when released from the grapple that removed it from the core. With the grapple clear, the mating half would be positioned. The figure shows the fuel loaded in the shroud. Note that end overlaps would

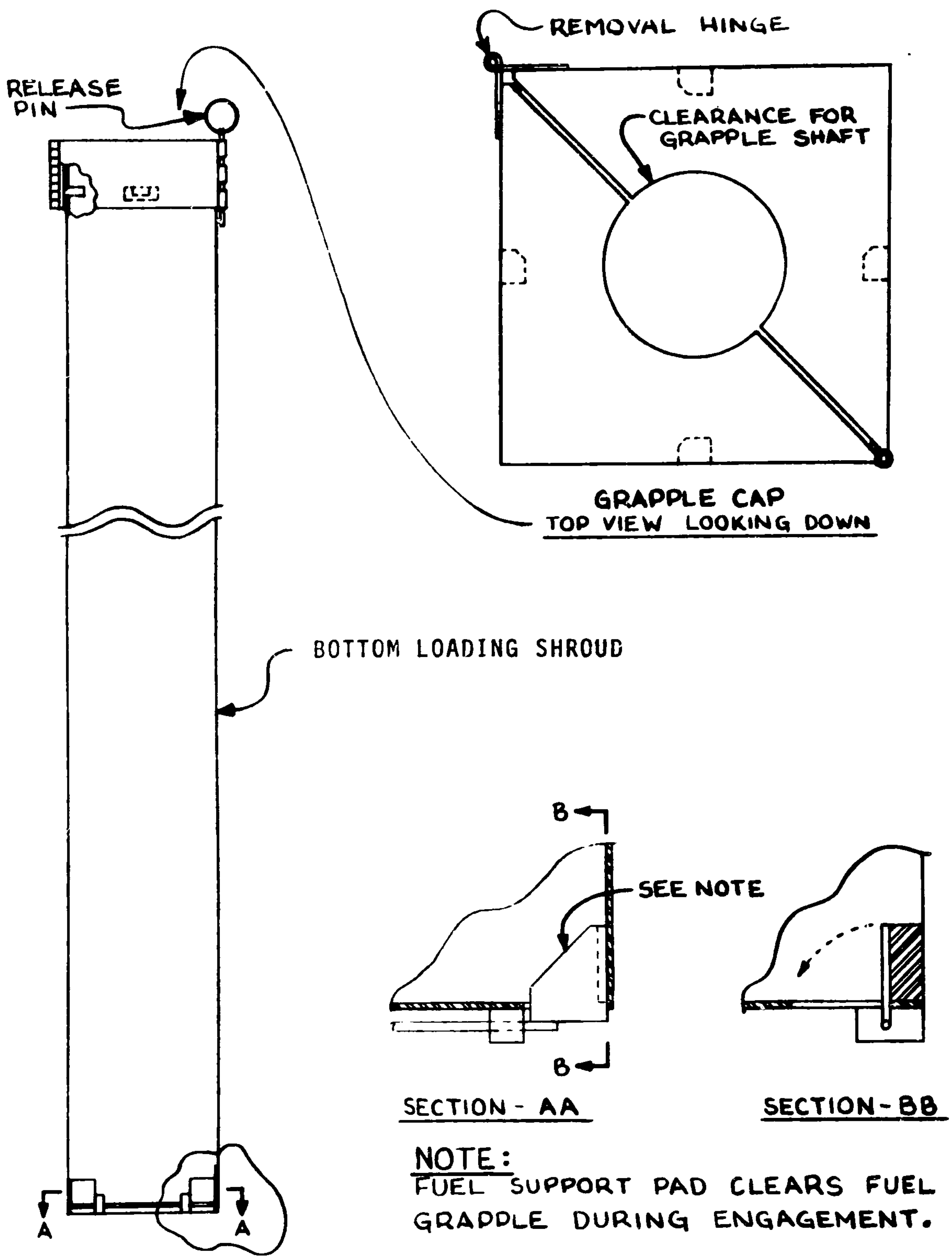

Figure 9. Bottom loading shroud concept.

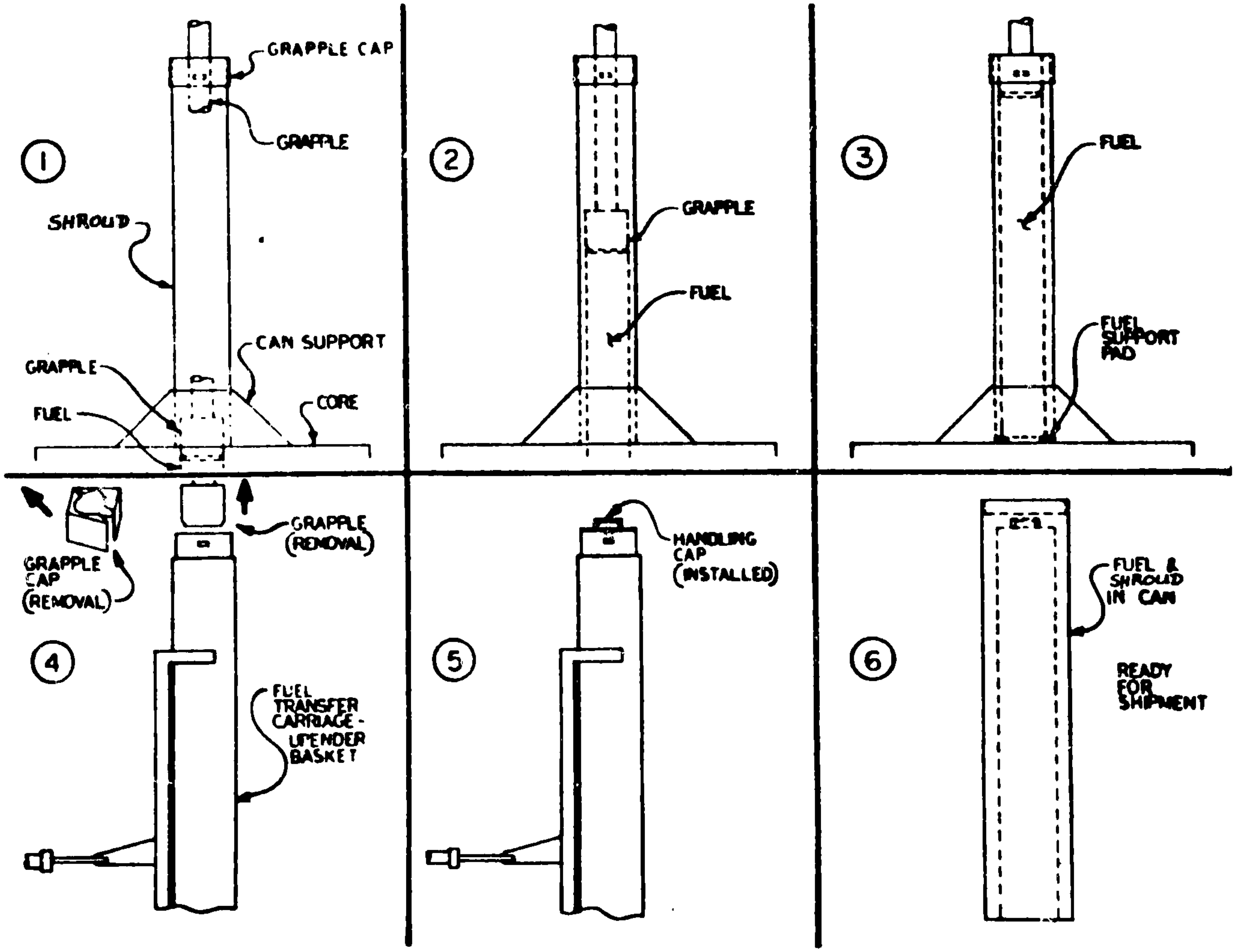

Figure 10. Shroud bottom-loading sequence.

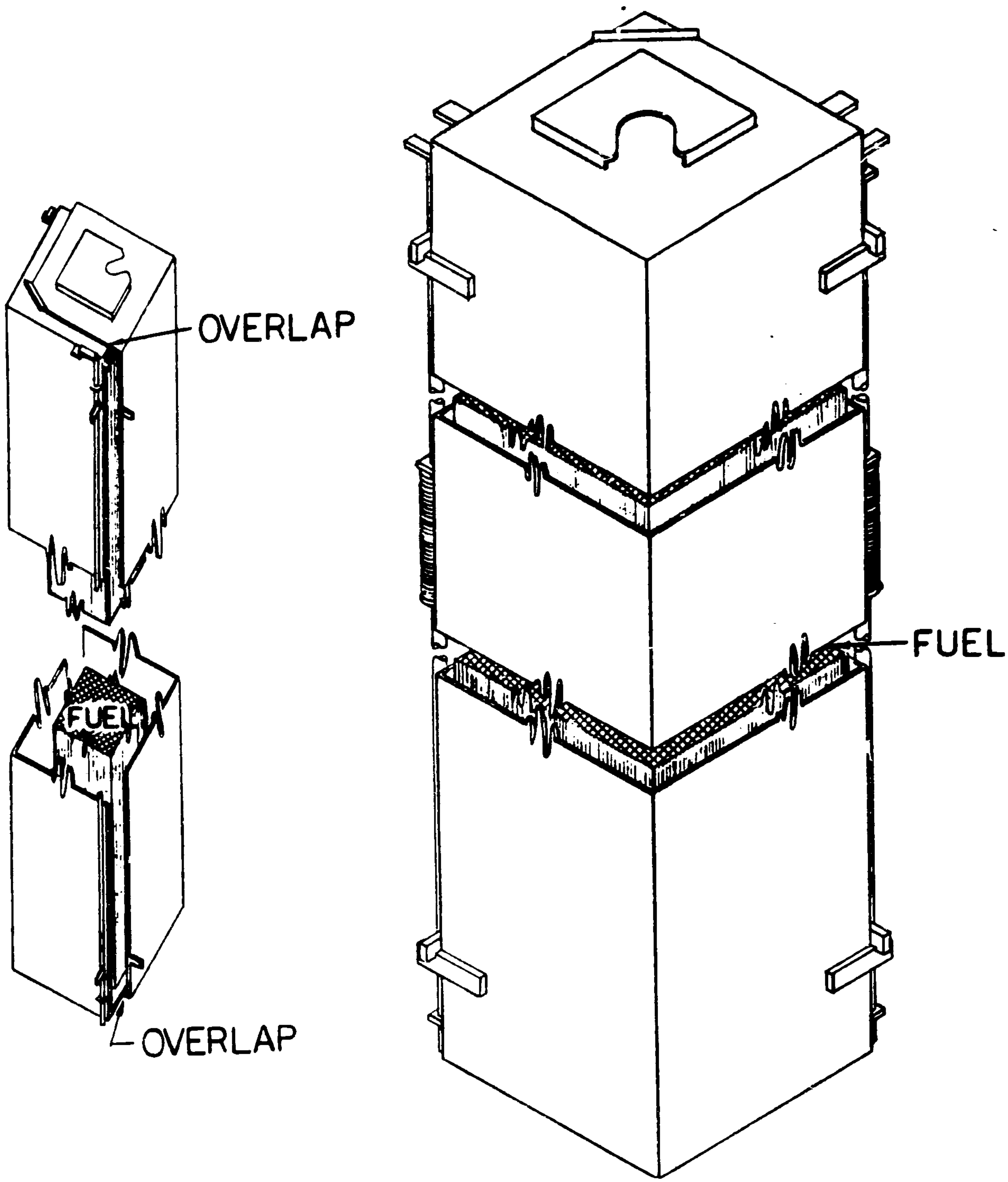

Figure 11. Loaded (two part) handling shroud concept.

prevent lengthwise displacement and resulting disengagement of the halves. Then spring-loaded latches would be remotely set. This would mechanically lock the halves together.

The shroud/canister approach could also be utilized for debris. Here, the shroud would be reconfigured to various special purpose baskets or filter cartridges or elements that would be filled by long handle, underwater tools, or by hydraulic suction. Debris would be defined as follows:

The deris might consist of:

- UO_2 pellets
- Cladding
- UO_2 powder
- Cladding fines
- Other fuel hardware fragments in a size spectrum ranging from visually discernible to fine particulate.

Debris would probably be located in the upper, central core region, or on the reactor vessel bottom (see Figure 12). Frozen core sections are not directly addressed but some form of remote sectioning would be assumed to render them dimensionally equivalent to either intact fuel or debris.

The first steps of the approach to can debris would be to utilize underwater tools (hook and tongs). They would be used to manually segregate and free, large-scale fragments into baskets for collection. Once the larger fragments were cleared, an underwater vacuum debris system would be placed on the core periphery. Alternatively, the vacuuming of fines could be performed prior to removing large fuel component pieces. The vacuum system would be operated remotely. The system would be provided with replaceable element(s) for both the larger and smaller sized particles.

After the baskets and replaceable elements are loaded with debris, they would be transferred and stacked within a shipping canister to ensure confinement of off-gas and particulate contamination. The same approach would be required for the shrouded fuel discussed earlier. This is shown in Figure 13.

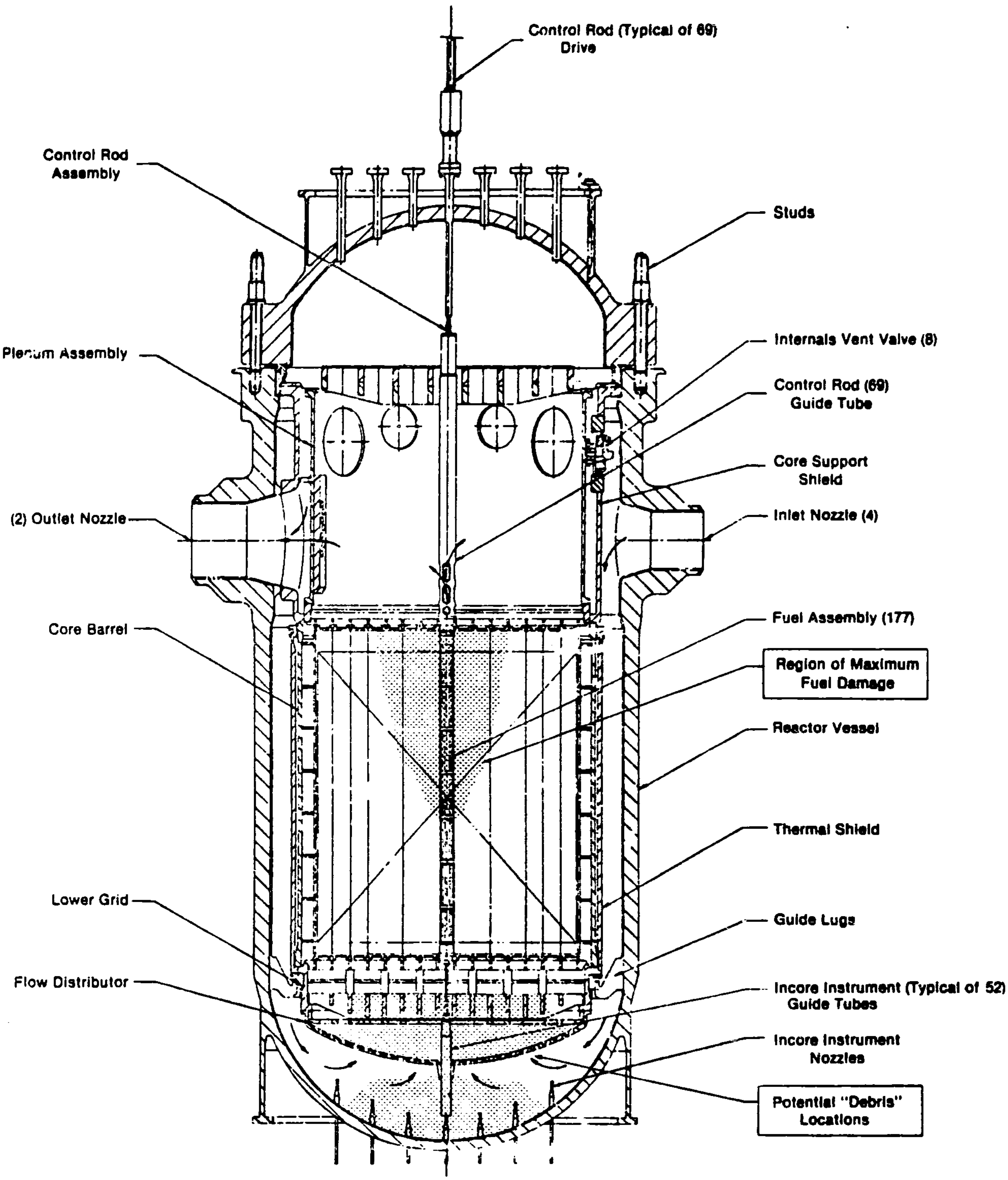

Figure 12. TMI-2 reactor vessel and internals-general arrangement.

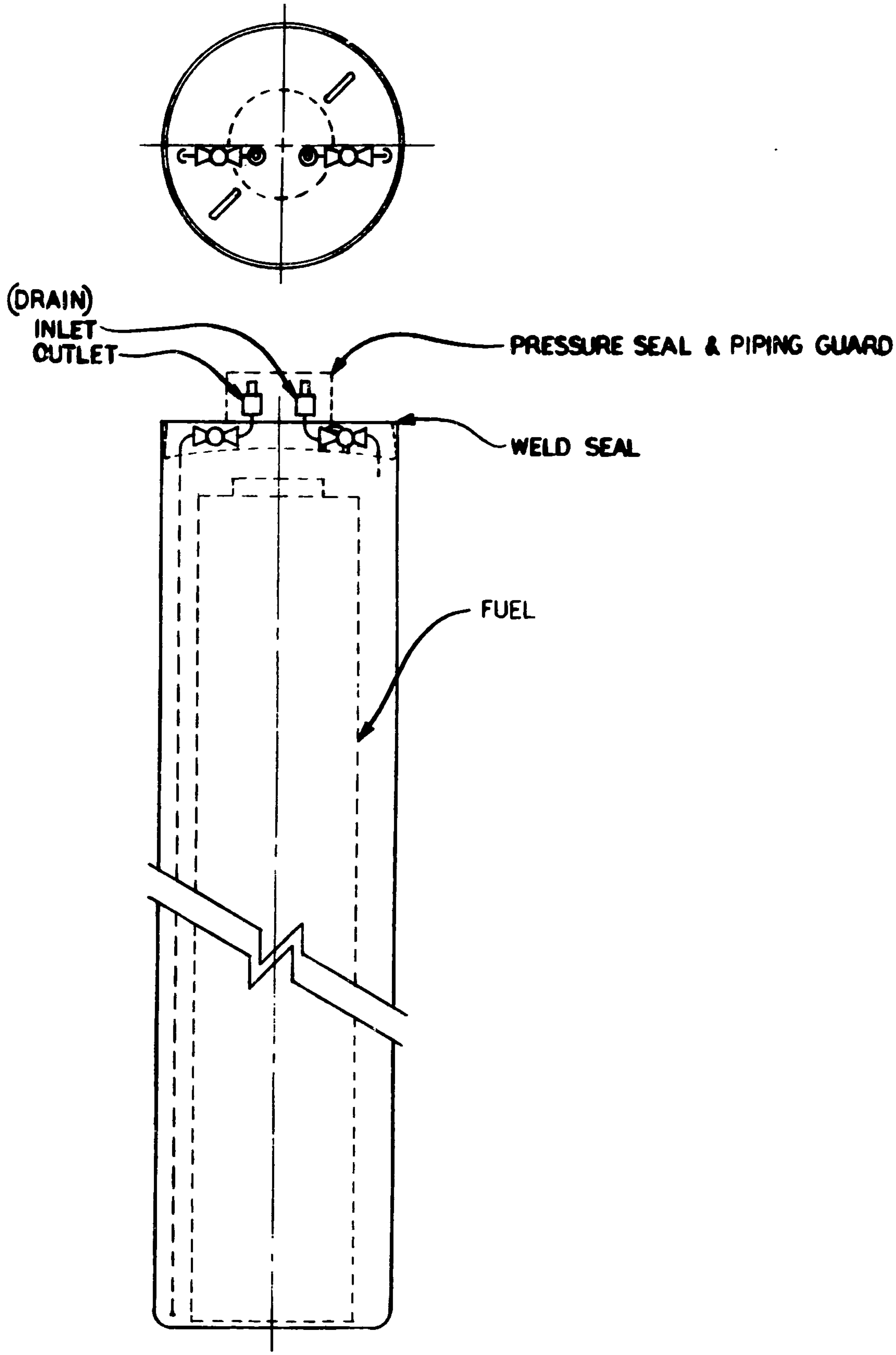

Figure 13. Pressure containing shipping canister (concept).

A drain for the canister would permit a pressurized draining of the canister interior. This in turn would drain both the shroud and fuel. All shroud concept designs should be either inherently free draining or should be provided with separate drains. However, fuel and canister displacement weights would preclude buoyant floating of drained and canned fuel. The drain system could consist of a drain leg and a pressure leg. The drain leg would be affixed to the canister cap and be the length of the canister. It would fit into the clearance between the canister cross section and the sidewall during cap installation. Both legs would be provided with remote valves and remote quick disconnects. This system would permit both connections to be on the canister upper lid.

The canister cap could be welded to form a pressure vessel. (Also, a valve pressure seal and piping guard could be necessary as shown [dotted] in the figure.) As an alternative, a mechanically secured cap is shown in Figure 14. This would facilitate unpackaging at a receiving facility if shipping requirements would accept the low internal pressure capability of this canister. Lower pressure retaining requirements are possible if it is confirmed most, or all, of the free fission gas was released during the accident. Also, the low decay level of the fuel results in less pressure retaining requirements.

The above discussion indicates that for a spectrum of fuel conditions and loading options, an inner shroud/outer shipping canister concept is possible. However, the approach places considerable emphasis on respect for the questionable structural integrity of the intact fuel. Since the initial lifting and interface with adjacent fuel may produce stresses that exceed those experienced during the canning operation itself, this approach may be overly conservative. It also requires an increased inventory of canning equipment and more involved canning operations.

Alternate Selection Criteria

The classical selection criteria appear to be as appropriate in the selection of a canning system for disposition of TMI-2 fuel as with less esoteric design problems.

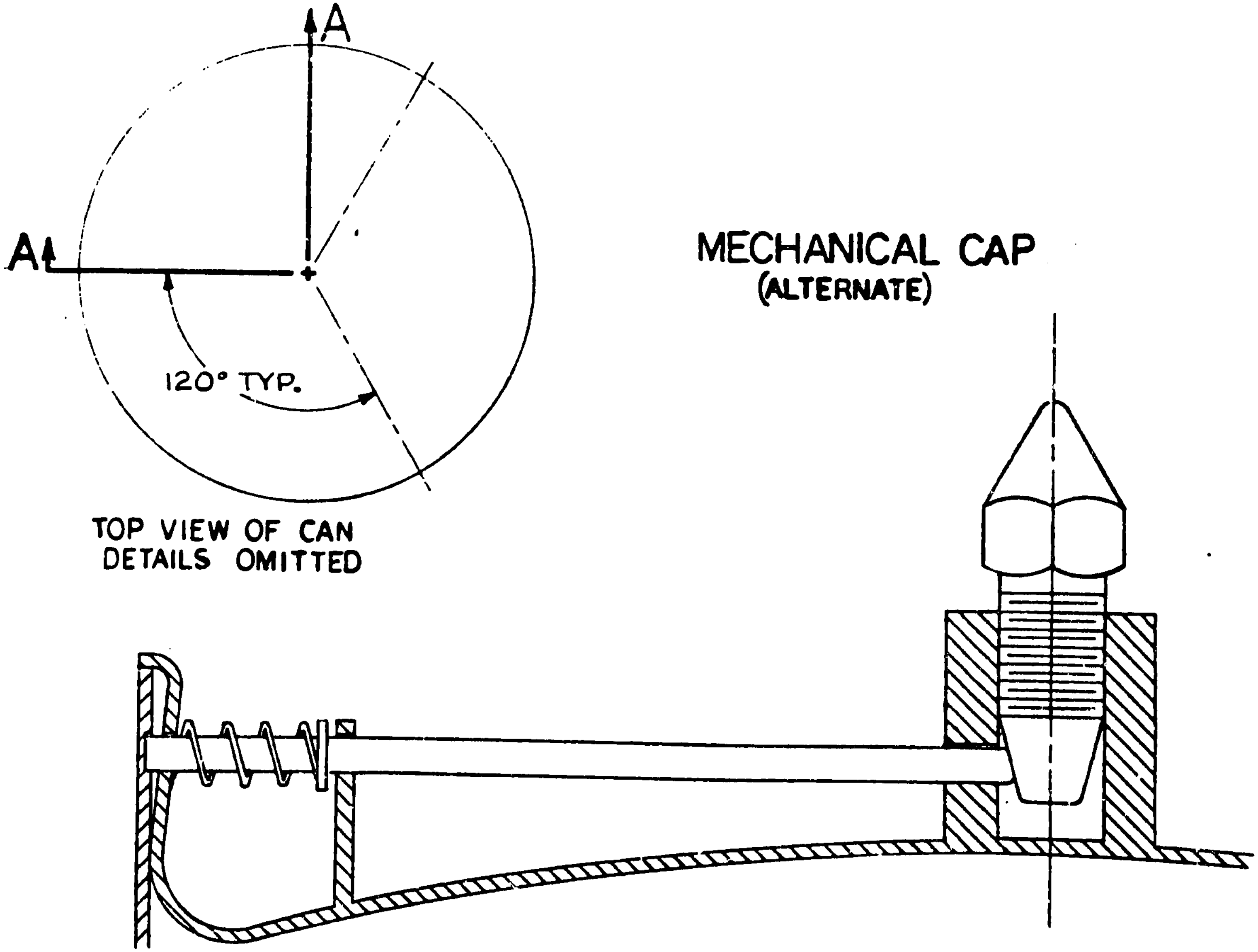

Figure 14. Mechanically sealing cap for shipping vessel closure.

So long as a thorough definition of the problem based on the available data does not preclude a simple, single solution, then this approach should be selected, particularly if it bounds a concensus viewpoint of the unknowns.

Such is the case with a single, multi-application canister. The spectrum of core conditions can be accommodated with this canister and operational conditions imposed by the radiological environment do not preclude its use (whether it be square or round in cross section). It is compatible with ALARA considerations from the viewpoint that an alternative which permits simpler training and operation as well as minimizes equipment transfers across radiation boundaries should result in lowered exposures.

The single canister for multi-application indicates lower costs from an inventory, a fuel rack and a manufacturing setup point of view.

Since a limited number of well defined categories of fuel condition do not exist, a variety of single-purpose canisters does not appear justified. Furthermore, fixed loading methods and disposition options are not available to justify such a variety of single-purpose canisters, particularly since a <u>single</u> multi-purpose application canister has been discussed in the preceding section that covers the potential fuel, loading, and disposition spectra. Selection of a variety of single-purpose canisters would also be expected to increase cost, increase tooling, require additional training, and be more involved to operate. Therefore, even though two alternatives for the single, multi-application canisters are presented in the following, the intent is that selection occur before detailed design is initiated so that, in fact, only a single, multi-application canister system is utilized.

The primary motivation for an inner shroud/outer shipping canister system is premised on minimizing fuel stress after core removal. Until it can be demonstrated that the lifting and adjacent fuel interface stresses are <u>less than</u> the canister loading stresses, this approach, for the reasons discussed above, is not justified.

Once the single, multi-application canister approach is selected, detail development is necessary to ensure that secondary considerations do not preclude its use. This development is provided in the following.

Single Multi-Application Canister Design Development

Capacity Criteria

As discussed earlier, the canister (whether it be square or round in cross section) must accommodate a fuel spectrum that includes the intact-fuel-with-nonfuel-bearing-component envelop. It must also accommodate a full load of debris. The envelop sets a length and cross-section minimum as noted below.

	Notes
Normal Length: 165 5/8"	
Maximum Length: 171"	Includes allowances for irradiation growth, thermal expansion at 212°F, and added length of nonfuel-bearing component.
Normal Cross Section: 8.536"	1/8" end fitting chamfers provide additional diagonal clearance.

Canister design weight calculations assumed the can alternative cross sections to be filled with UO_2 powder and the internal cavities to be those of the final concept designs as shown in Figure 15 and 16. The resulting design weight was conservatively set at 6000 pounds for the square alternative and at 8500 pounds for the round alternative.

Configuration

The configuration of the canister as shown in Figure 15 is square which accommodates the cross section of the fuel with a water reflected, critically safe geometry. A cylindrical can is shown in Figure 16 which does require

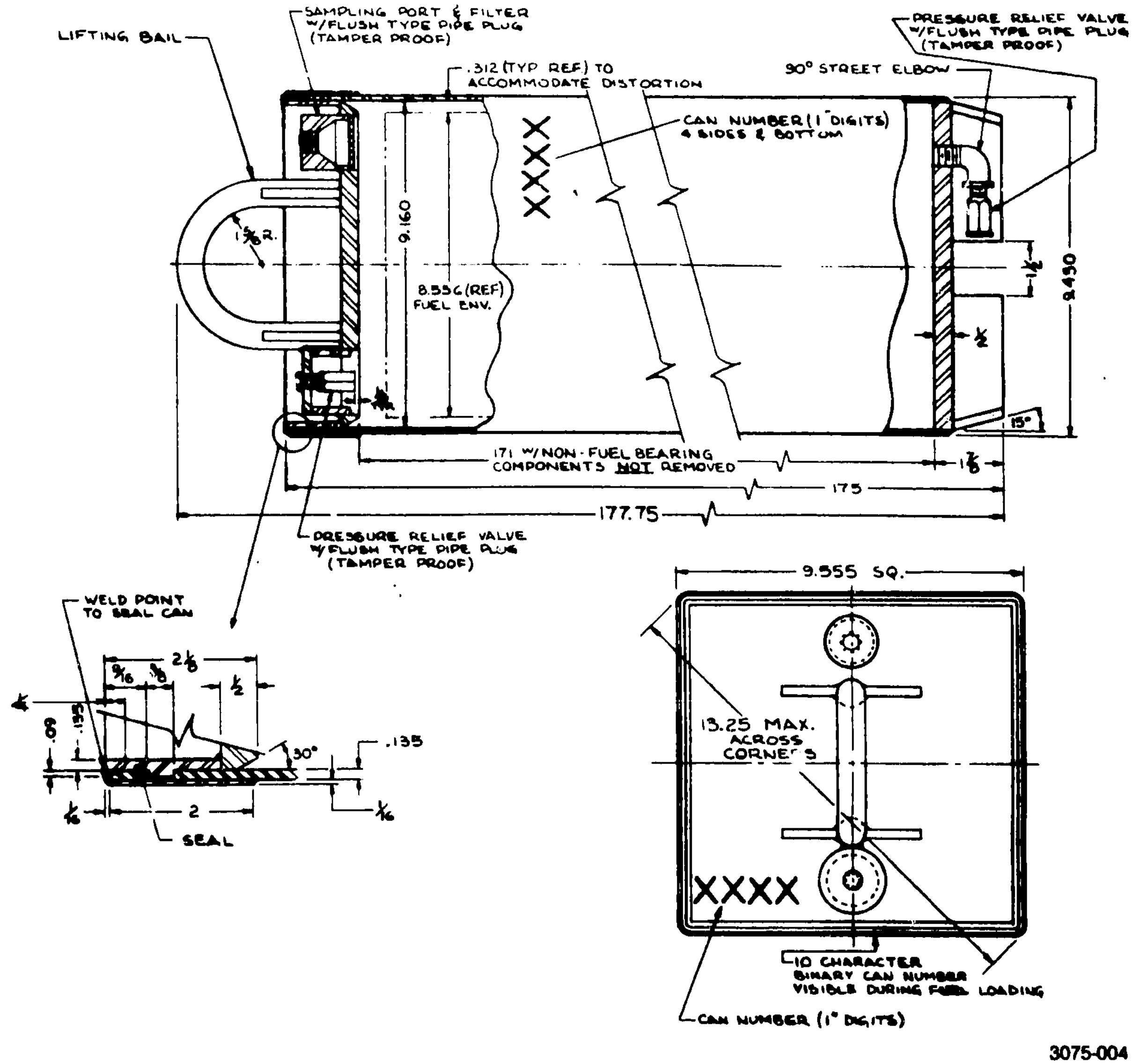

Figure 15. Single multi-application square canister.

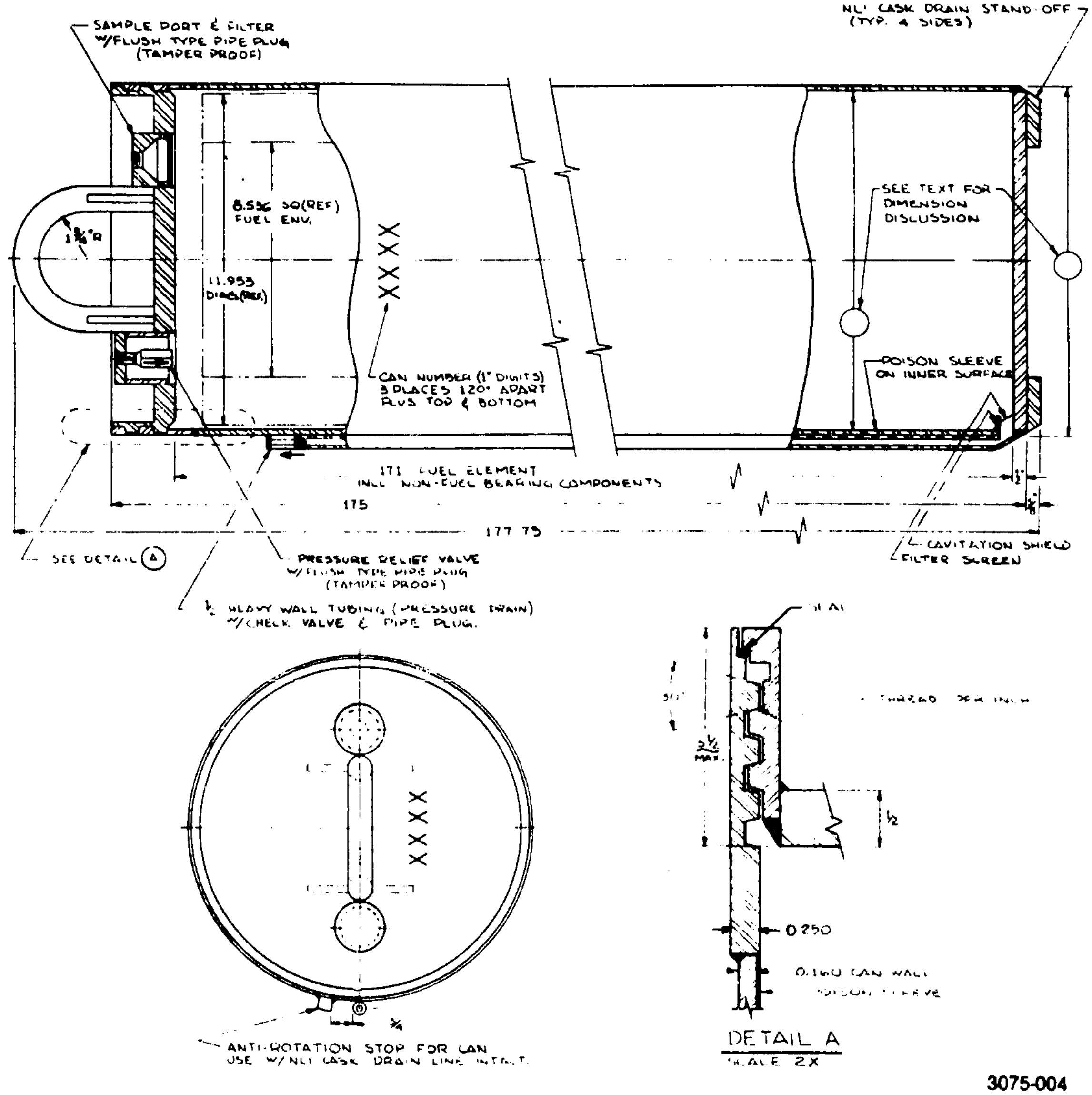

Figure 16. Single multi-application round canister.

poison but which provides an improved end view loading area which lessens the potential for interference with the damaged fuel during loading.

The canister and cap for both alternatives are provided with flushing, vent, and sampling fittings. Because canister loading operations will take place underwater, means should be provided to remove entrapped water from the loaded canister. A simple way to do this is to purge the canister with pressurized gas (e.g., air). Canister purging can be done by the purge gas entering at the top of the canister and venting at the bottom. As shown on the round canister, the drain could be routed up the side of the canister to facilitate hook-up to both purge and drain points with the canister in the vertical position. However, the option exists to vent within the refueling canal without hook-up or to use the upender for access to a bottom drain. After the water is purged, the check valves will prevent water from reentering the canister. Drain line pluggage may indicate redundant drains. If a permanent seal is desired, the outlets to the purging, venting, and the sampling connections could be plugged with luted pipe plugs. The bail for the square canister affixed to its cap is purposely oriented as shown to permit "walking" the canister from vertical to horizontal and vice versa. Its orientation precludes rotation of the square canister as could be expected from a diagonally oriented bail. In both cases, the relative positions of the bails and the flush, vent, and sampling fittings on the caps precludes lifting hook damage to these fittings during "tilting."

Both cap designs and the square alternative bottom design provides protecting structures to minimize damage to vent and sample fittings. The bottom structures are further configured to provide remote lead-in to holding stands or storage racks. Similar lead-in surfaces are provided on the caps to facilitate makeup with the canisters. The slots in the bottom structure of the square canister and the shim blocks on the bottom of the round canister preclude interference with the NLI 1/2 transport cask internal drain line.

The square canister cap has raised projections on its four sides that fit with inside slots on the canister upper walls when positioned to close the canister. The square canister wall and the cap sides are elastically strained during this engagement. By design, the square canister cap projections preclude removal or further insertion after closing. "O" ring temporary sealing for purging is provided. However, if a demonstration indicates "O" ring "roll out," below boiling fusible inserts could replace the "O" ring. However, additional tooling (i.e., heat source) and handling could be involved in the use of fusible inserts. Handling of an empty square canister will be implemented by a grapple that secures itself to the projections provided by the outside top band (which seals the capping slots) or the inside capping slots.

The round canister cap is secured wtih a modified Acme thread to prevent loading damage. The thread should also have a "Higbee" thread start (i.e., square and not tapered) to minimize starting damage. Tapered remote lead-in is provided with a straight "thread-alignment" lead-in following. "O" ring temporary sealing is provided for purging. Non-galling material such as Nitronic 60 should be used on one side of the thread interface. The thread interface is separately machined and then welded to the canister body to preclude essentially unattainable canister tolerances. The threads provide handling tool attachment means.

The overall configurations were also selected to minimize decontamination of the canister.

Shipping Cask and Upender Interfaces

The dimensions shown on Figures 15 and 16 result from considerations that include shipping cask and transfer canal upender (fuel transfer carriage) interfaces.

Figure 17 shows limiting shipping cask cavity dimensions. The overall canister plus bail length of 177 3/4 inches is chosen to permit thermal expansion without exceeding the cavity constraints of the NAC-1 which is limiting. Likewise, the 175-inch canister length is chosen to permit thermal expansion without exceeding the NLI-1/2 cavity constraints. The square canister diagonal dimension is chosen to provide 1/8-inch minimum clearance to the NLI-1/2 13.375 inches cavity at the top band. Insertion clearance is assured by the 1/4-inch maximum clearance over the rest of the square canister. Additional clearance which could generate undue impact during shipping must be addressed if utilized.

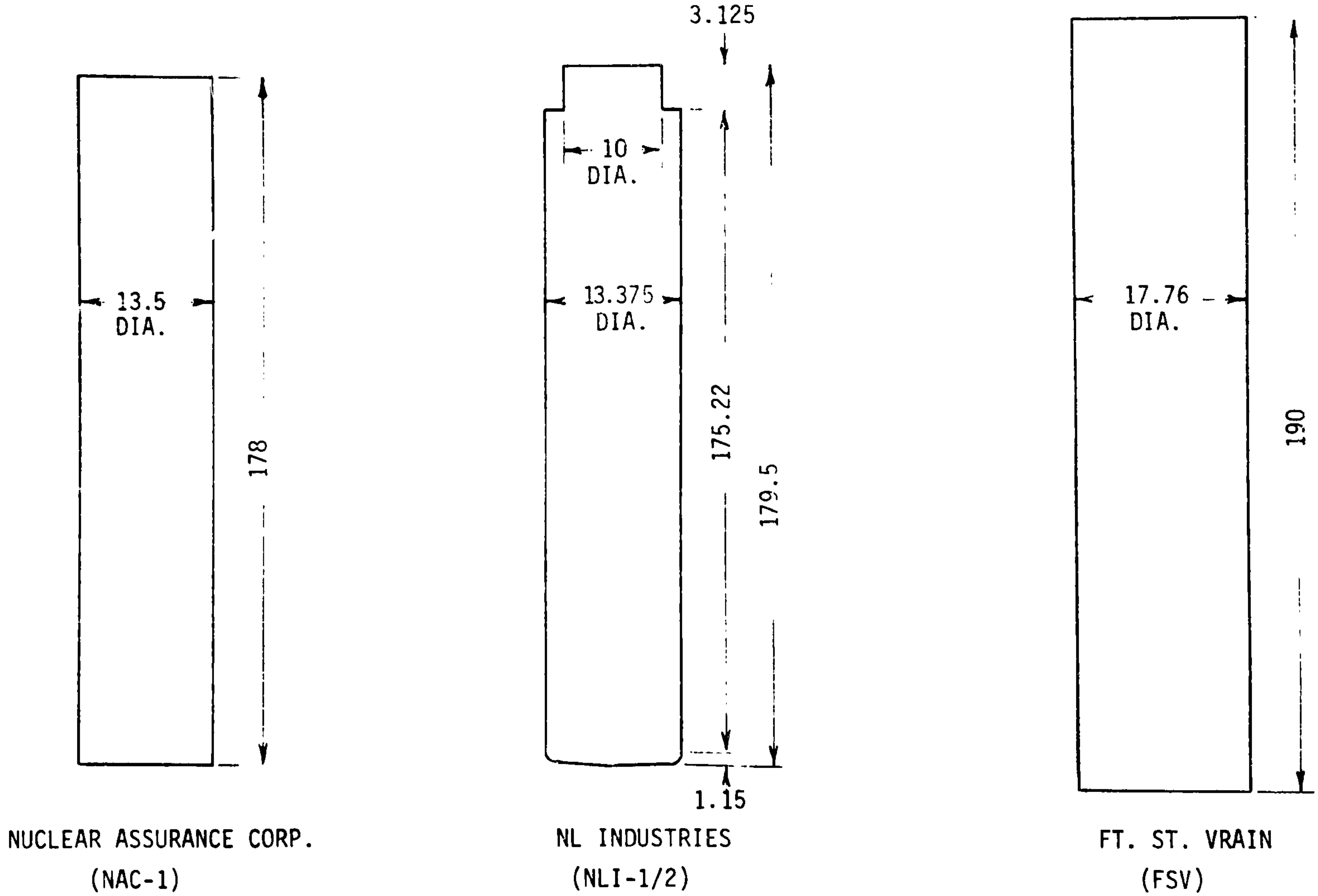

Figure 17. Truck cask cavity dimensions (inches).

Table 2 is provided to clarify the discussion of round canister cask interfaces as well as to indicate "trade-offs" and options. The cask compatibility of the five cases is shown in Column 2. The "All (legal weight truck vs. overweight truck) LWT" entry still assumes modification of the upper portion of the NLI drain line as discussed in the cask "pro and con" discussion that follows. Case IV assumes the NLI drain line is removed and cask rotation to drain which is also discussed below. Only Case V limits cask use to the FSV cask.

Obviously, Case I is prohibitively expensive and the fuel-to-can clearance is tight even though other considerations are acceptable. As discussed in the Round Canister Criticality section, K_{eff}'s of these concept options greater than 0.93 lessen the likelihood of ultimate licensing even though permitting desirable mechanical options.

With the preceding cask clearance discussions in mind, the following discussion on the desirability of the various casks relative to each other must be considered.

Six types of spent fuel transportation casks potentially are available for use in moving TMI-2 fuel from that site. None of the six types are now certified to haul damaged fuel. The certification process undoubtedly can be accomplished more readily for some models than for others, and the degree of assurance of certification should be the paramount parameter in the selection of a cask. In this light, truck casks are better candidates than rail casks; legal weight truck casks are likely to be preferred over overweight truck casks--purely from the viewpoint of ease of certification.

Following is a summary of the pro and con attributes of each cask system (except criticality, see page 56).

NLI 1/2 (LWT). This cask is relatively easy to operate. It has the best remotely operating lifting yoke of all potential casks. It probably would be the best candidate for a license amendment for hauling damaged fuel, so long as no modifications are made that would affect the primary pressure boundary, for the following two reasons. It is the most recently licensed LWR-LWT cask having been subjected to the more stringent licensing questions of recent times. Also, its dimensions preclude multiple canister loadings which reduce cask

TABLE 2. ROUND CAN INTERFACE CONSIDERATIONS

Case	Cask Compatibility (1/4" Minimum)	Poison Sleeve	Poison Sleeve Thickness	Unit Poison Cost Estimates	Total Canister Cost	K_{eff}[b]	Nominal Fuel-To-Canister Clearance	Nominal Canister Outside Diameter	Side Drain	Case (see pp 51 & 52)
I	ALL LWT	SS/AL Clad B_4C (91% Enriched B_{10})	0.080	$40,000	41,650	0.929	3/16"	12 5/8"	Yes	E
II	ALL LWT	SS Clad Cadmium	0.080	$ 2,000	3,650	0.940	3/16"	12 5/8"	Yes	H
III	ALL LWT	1% Borated Stainless Steel Can	N.A.	$ 1,350	3,000	0.946	3/16"	12 5/8"	Yes	D
IV	ALL LWT[a]	SS/AL Clad B_4C (Boral)	0.200	$ 1,750	3,400	0.925	5/16"	13	No (Bottom)	F
V	FSV	SS/AL Clad B_4C (Boral)	0.200	$ 1,750	3,400	0.925	5/16"	13	Yes	F

a. NLI cask drain line removed necessitating cask rotation fixture for cask draining.

b. See p. 53 underlined phrase. These K_{eff}'s are based on water reflection. See the Cask Shipping Criticality section for representative values in this nonwater reflected environment.

poison questions so long as the individual canisters are sufficiently subcritical.

On the other hand, the cask has a metallic primary seal that is easily damaged during remote head placement (a TV camera at pool floor elevation plus 17 feet could help alleviate this). The impact limiters could be expected to provide some operational difficulties even though these would not be insurmountable. Also, all primary containment penetrations are through the inner head and this necessitates a drain line along the entire length of the cavity unless a fixture is provided to rotate the cask horizontally for draining. Problems can be avoided if the canister is designed with this in mind. Even if the canister is designed to accommodate the drain line, the upper portion of the line, as well as its mating member on the inner head, would require modification that appears feasible to accommodate the required canister length.

Five of these units are in existence and this should be sufficient to handle continuous (16-hour loading cycle) shipments to any point within 800 miles of TMI.

NAC-1 (LWT). Operations of this cask is the simplest of all available casks (e.g., bottom free drain). Certification of this cask for new application might be a problem based on recent restrictions attached to the current certifications. These restrictions appear to be based on NRC concerns for the thermal and resulting structural limits of this cask which cloud the prospects for future licensing efforts. The yoke "lay-down" technique, as opposed to disengagement from the cask underwater, may require more horizontal crane travel than can be accommodated without cables touching the loading pool curb when the cask is at the bottom of the pool. Potential use of this cask should not be assumed until clearances have been physically verified as acceptable, using an on-site run. Also, only three of these casks are available today with one doubtful for future work; not enough for an uninterrupted shipping shedule.

FSV-1 (LWT). The cask has the largest cavity (17-3/4 ID x 190 inches L) of all LWT casks. It was designed for HTGR spent fuel and has never been certified for LWR fuel, so it probably would have a time-consuming certification period. None of the three existing casks is available full time (PSC of Colorado has first priority at all times). If this cask were to be used, about five new ones should be built. The new model should be shorter 10 inches to save 2000 pounds in weight which would probably be necessary to remain an LWT. Development work would be needed to qualify this "dry" loading cask to underwater use.

TN-9 (mod) (OWT). The multiple cavity TN-9 is being modified so that one available option will provide a single 20 to 24-inches inside diameter cavity about 180 inches long. If it was then desired to haul multiple canisters of TMI damaged spent fuel in this cask, the criticality control would be a difficult licensing consideration. It probably could be handled with extensive basket poisoning. If it could be certified to haul multiple canisters of damaged fuel, improved shipping rates could be anticipated. The remotely operated lifting yoke for this cask has not been demonstrated in the U. S. to the knowledge of AGNS and based on the experience of others using similar equipment, it appears prudent to request such a demonstration. If a shelf is to be built in the loading pool, strength requirements of the structure and consequences of a drop accident will both increase above that required for LWT.

IF-300 (Rail). It would appear that this cask model was the one for which the TMI-2 loading pool was designed. However, the two have sufficient incompatibilities such that selection of this cask should not be made prior to extensive on-site testing. A special head handling gantry probably would need to be provided.

NLI 10/24 (Rail). This cask is too big physically to be handled safely in the TMI-2 loading pool. The cask is untried in actual service.

In summary, the NLI 1/2 is the preferred cask for this job. None of the casks should be completely ruled out, except for specific obstacles to certification, or mechanical difficulties that may arise as detailed evaluations proceed.

An evaluation of the interfaces between canned fuel assemblies and the fuel transfer carriage basket of the upender was performed by reference to Babcock and Wilcox drawings of the fuel transfer system: 44-54-009-03, 44-54-011-02, 44-54-017-08, 44-54-040-30.

The evaluation was used to determine that the limiting dimensional interfaces were those with the available shipping cask and not those with the upender.

The maximum canned fuel length that could be accommodated by the carriage basket was investigated. Assuming the stop assembly (Item 1170, 44-54-011-02) could be relocated, the limiting interference would be the flange of the transfer tube (Item 3367, 44-54-040-30) on the reactor side. Use of this maximum length might require removal of the pool emergency cable system (Item 0022, 44-54-017-08), but insufficient detail was available to say with certainty. The maximum, uninterfered length would be 15 feet 1/8 inch. The maximum square cross section that could be contained within the 15.25-inch octagonal inside diameter of the basket would be 11.48 inches. Thus the NLI cask inside diameter of 13.375 inches and the 178 inches length of the NAC cask cavity are more limiting. Once dimensions were fixed, canister wall thickness was investigated to establish internal cavity size.

Wall Thicknesses

The basic wall thickness was assumed to be contingent on whichever was the controlling case between tension stress that develops in a vertical lift and bending stresses that develop in "tilting" the canister (modeled as a box beam) from horizontal to vertical or vice versa. As discussed later, these developed thicknesses were checked against various internal and external pressurizations which generally need not be considered together with the above structural stresses, since venting is permitted. Combination of stresses were deferred as final design considerations since safety factors or margins to allowables were generally conservative and the necessary combinations of routine and accident conditions with typical adjustments of allowables were not defined. Based on the material selection of AISI 304L or 316L stainless steel as discussed in the Material section below, an allowable stress of 10,000 pis was chosen to to provide a three to one safety factor in this and other nonaccident structural calculations for the canister designs that are dynamic in nature as opposed to static loading under pressurization. The bending stresses developed in "tilting" were shown to be controlling and fixed the wall thickness at 0.135 inch for the square canister and at 0.160 inch for the round canister.

The bottom plate and cap thicknesses were calculated to provide a three to one safety on yield using the 6000 pound design load for the square canister and the 8500 pound design load for the round canister distributed as a uniform pressure across these surfaces.

Accident conditions were analyzed using loadings developed from a selected cask analysis where lengthwise loading was assumed at 10 G's, 5 G's laterally, and 2 G's vertically. The allowable loading was revised to 0.9 of ultimate strength or 72,000 psi (i.e., deformation was allowed but failure was precluded). The selected thicknesses were shown to be adequate.

With the wall thickness fixed, the internal cavity of the square canister was established as 9.16-inch square. This included a corner inside radius of 1/8 inch which matches the 1/8 inch x 1/8 inch chamfer on the fuel assembly end fitting edges. This allows a total of approximately 5/8 inch in excess of design fuel cross-section to accommodate fuel distortion resulting from the accident.

As shown in the "Nominal Fuel-to-Canister Clearance" column of Table 2, these clearances for the round canister were either fixed at 3/16 or 5/16 inches to the intact fuel diagonal of 11.953 inches that takes credit for the end fitting 1/8-inch chambers. As discussed in the Round Canister Criticality section additional clearances are not promising.

These cavities were next investigated for "worst-case" criticality considerations.

Criticality Safety

A complete assessment of canister criticality safety requires an evaluation of canning, handling, shipping, and storage. It is not within the scope of this study to perform a complete examination of all of these areas; rather, several key analyses were performed to determine the configurations and conditions where criticality safety may be a limiting condition and to help define future directions in the canister design process. One of the fundamental configurations of interest is a single canister immersed in water, and this was examined for both the square and round canister. Next, considerations were given to an array of canisters as would exist in a storage environment and to a representative shipping cask environment.

Square Canister. The 9.16-inch square fuel canister is acceptable for TMI-2 fuel from the standpoint of nuclear criticality safety in a water reflected environment. This conclusion is based on: (1) information available in the literature that shows safe dimensions for low enriched uranium fuels, and (2) the results of a criticality analysis of the filled canister.

For these evaluations and those that follow for the round canister, the following "worst-case" conditions were assumed:

(1) No credit was taken for fuel burnup.

(2) All fuel in the canister was assumed to be at the highest enrichment for the TMI-2 fuel (i.e., 2.96% enrichment).

(3) No credit was taken for cladding or structural materials that may be present.

(4) No credit was taken for poison materials that may be present (e.g., soluble boron in solution, burnable poisons, nor control rod materials.)

(5) The worst-case form of UO_2 fuel was assumed (i.e., full-size fuel pellets in a unborated water medium at 70°F).

(6) The worst-case ratio of the volume fraction of fuel to the volume fraction of water in the canister was assumed (i.e., optimum moderation).

(7) The canister was assumed to be surrounded by a water reflector at 70°F containing no soluble poison. As opposed to reactor storage pools,

away-from-reactor or reprocessing storage pools are not typically borated and to assume otherwise would limit dispositon options.

These "worst-case" conditions correspond to a condition where uncladded loose fuel pellets of the highest enrichment fill the canister. Other conditions such as cladded fuel rods or whole fuel assemblies would be less reactive.

The basic square canister design consists of a square 9.160 inches on a side (inside dimension). Its length of 170+ inches is essentialy infinite from the standpoint of criticality. A 9.160 inch square corresponds to a cross-sectional area of 83.906 square inches. If this area is assumed to take on a circular shape (wich is slightly more reactive than the square shape), the resulting diameter is 10.34 inches. The Nuclear Safety Guide,[15] Figure 2-15, shows that an individual cylinder containing 2.96% enriched uranium in the oxide form is "safe" (i.e., subcritical) provided that the diameter is $\leq$10.79 inches.[a]

On the basis of the above values, it is concluded that the square canister with an effective diameter of 10.34 inches is less reactive than the "safe" cylinder diameter of 10.79 inches.[15]

As a further check on the criticality safety of the canister, a specific criticality analysis was made of a filled container. Uncladded UO_2 pellets at 2.96% enrichment and water at the optimum pellet-to-water ratio were assumed to exist in an infinitely tall canister 9.16 inches square. The UO_2 pellets were assumed to have a density of 92.5% of the theoretical value. The analysis employed the NITAWL/XSDRNPM/KENEO-IV computer program with the XSDRN 123 energy group neutron cross-section set. Results shows a K_{eff} of 0.944 $\pm$ 0.011[b] for the case involving full water reflection. An identical calculation representing the square canister as a cylinder instead of a square (while preserving the crosssectional area) showed very similar results, K_{eff} = 0.947 $\pm$ 0.013.

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a. The "safe" cylinder diameter of 10.79 inches corresponds to an estimated K_{eff} = 0.98. The estimated critical dimension (i.e., K_{eff} = 1.00) is 11.34 inches. It is also noted that the Reference 15 uses the theoretical density of UO_2 in its evaluations, which is conservative.

b. Two standard deviations (i.e., 95% confidence level).

Based on the above results, it is concluded that the square canister design is satisfactory in a water-reflected environment from the standpoint of nuclear criticality safety. Two points of caution are noted, however. First, the K_{eff} is quite sensitive to the exact dimensions, and the value of 9.160 inches should be considered as the upper limit when including allowance for manufacturing or "bulging" tolerances. Secondly, the evaluation discussed above applies to a single, isolated canister. Additional criticality safety analyses would be required for any case in which a filled square canister is in close proximity to other fuel material or canisters so as to cause neutron interaction or nonwater reflected environments.

Round Canister. The following criticality results substantiate the K_{eff} values shown on Table 2 above. After running bounding cases for the K_{eff} of the unpoisoned and "infinitely" or "maximum" poisoned round canisters, the K_{eff} of various candidate poisons was analyzed. Generally, the approach involved determining, within mechanical constraints, K_{eff}'s of various selected poison systems.

It should also be noted that intact fuel-to-can clearances greater than those utilized in these calculations (either 3/16 or 5/16 inches) and their resulting can I.D.'s will significantly alter the resulting K_{eff}'s (approximately, 1% ΔK for each 1/4 inch ΔI.D.) which are, of course, based on canister loadings of UO_2 pellets and water as discussed below. Also, since for the various poisons studied, the apparent point of "diminishing returns" was being reached, relative to the amount of poison utilized, little opportunity exists to "poison away" ΔK increases due to increases based on debris in these intact fuel clearances unless generally accepted criticality margins can be justifiably decreased.

Background and Discussion--Conceptual design criticality calculations have been performed to investigate the round canister concept for canning TMI-2 fuel.

The canister is to be used for TMI-2 fuel in any form; for example, intact fuel assemblies, loose fuel pellets, and debris. As required for criticality safety reasons, a neutron poison may be incorporated within the canister wall (e.g., borated stainless) or in a thin layer on the inside of the container wall (such as cadmium or a boral sheet).

The dimensional restraints on the ID and on the OD are "tight." The canister ID must be large enough to accommodate an intact fuel assembly, and the canister OD is limited by the inside dimensions of available shipping casks.

Assumptions and Methods--The canister was assumed to be filled with UO_2 pellets and water, and no credit was taken for cladding, structural materials, poison rods, nor soluble poisons that may be present. The enrichment was 2.96%, which is the highest initial enrichment for TMI-2 fuel. No credit was taken for fuel burnup. All materials were assumed to be at room temperature (i.e., 70°F).

In the primary series of calculations (reported in Results - Single Canister in Water section below) a value for the fuel pellet-to-water volume ratio was chosen that is known to be the optimum (i.e., highest K_{eff}) for a system containing only pellets and water (i.e., no peripheral poisons). It is possible that the existence of a peripheral neutron poison could cause a slight shift in the optimum pellet-to-water ratio and, thus, increase the system K_{eff}'s above the values reported. This effect was addressed and is discussed in Effects on the Water-to-Fuel Pellet Ratio on the K_{eff} section below.

All calculations reported herein employed the NITAWL/XSDRNPM/ KENO-IV computer programs with the XSDRN 123 energy groups cross-section set. KENO-IV is a MONTE CARLO program, and the results are subject to statistical variations. All reported KENO results herein are shown with two standard deviations. (i.e., 95% confidence level).

Results - Single Canister in Water--This section deals with the K_{eff} of a single canister in water.[a] Comments on the criticality consideration of an array of canisters, as in a storage environment, will be discussed in An Array of Canisters section below.

a. In these calculations, water fills the space within the fuel region that is not occupied by UO_2. There is an "infinite" water reflector surrounding the canister.

A total of seven cases was run to examine the K_{eff} of a single canister in water. In these cases, the type and amount of neutron posion material associated with the canister were varied. There were also slight variations in the basic canister dimensions. These cases are described as follows:

Case A

This case assumes a model of a stainless steel canister with an ID of 12.288" and an OD of 12.625". There are no special neutron poisons (e.g., boron or cadmium) associated with the system. The calculated K_{eff} is 1.012 ± .016.

Case B

Case B assumes the use of borated stainless steel for the canister wall with a natural boron content of 0.18% by weight. The canister dimensions are the same as Case A. The effective areal density is 0.0011 grams B-10/cm^2.[a] The calculated K_{eff} = .963 ± .016.

Case C

This case is identical to Case B except that the content of natural boron was raised to 1.08%. This gives an areal density of 0.0067 gram B-10/cm^2. The calculated K_{eff} = 0.944 ± .012.

Case D

Case D is similar to Case C except that the boron content was lowered slightly from 1.08% to 1.00% and the cylinder ID was reduced from 12.288" to 12.141". The areal density of the container wall is 0.0088 grams B-10/cm^2. The calculated K_{eff} = 0.946 ± .012.

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a. The areal density is the density of B-10 (i.e., gram B-10/cm^2) in the poison material times the thickness of the material.

Case E

This case was run to examine the effect of significantly higher a real densities of boron. In this case, the areal density was raised to 0.0608. The calculated K_{eff} was 0.929 ± .014.

Case F

This case examined the use of boral. The stainless steel canister ID of this model[a] was placed at 12.578" and the OD was 12.898". An active poison layer of 0.155" thick boral was located on the inside of the canister in contact with the stainless steel wall. In this case, the areal density of B-10 was essentially identical to Case E (i.e., 0.0608 gram B-10/cm^2). The calculated K_{eff} = 0.925 ± .012.

Case G

This case examined the use of cadmium. The canister ID and OD of the model were 12.288" and 12.625", respectively. A layer of cadmium 0.005" thick was placed on the inside canister wall. The calculated K_{eff} = 0.946 ± .014.

Case H

This case is identical to Case G except a 0.05" thick cadmium sheet was located on the inside wall. The calculated K_{eff} = 0.940 ± 0.12.

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a. Here and in the following cases, inert materials that would be expected, in practice, to clad the poison are neglected. So long as the conceptual canister clear ID for fuel is equal to or less than those of the model, the results will be valid.

Case I

Case I is of interest in that it provides a theoretical lower limit on the K_{eff} of the system if an "infinite" amount of poison were included in the canister wall.[a] The calculated K_{eff} = 0.918 ± .012.

Figure 18 shows a plot of K_{eff} versus the areal density of boron. These results are taken from Cases A through F. Case I is also shown.

From the results in Figure 18, it is quite clear that special neutron poisoning materials are required to achieve an acceptable criticality safety posture. Depending on the amount of neutron poison included in the system, the K_{eff} will vary in the range from 1.01 to 0.92.

A generally acceptable design practice is to have a margin of safety of about 0.05 ΔK to a critical condition. On this basis, the system K_{eff} should not exceed 0.95 under worst-case conditions after accounting for statistical uncertainties[b] and for the "bias" in the calculational model. The calculational model "bias", which is properly a subject of detailed design, has not been determined to date for the conditions examined here. It is not inconceivable that a bias in the range 0-2% could exist indicating reasonable confidence only for K_{eff} <0.93 at this conceptual design stage. Based on these factors and the results in Figure 18, it is estimated that an areal density of at least 0.007 and possibly as high as 0.06 will be required for the boron-10. A real densities in this range can be achieved with a boral plate.

A cladded cadmium sheet can not be ruled out as a possible poison material; but, even at 0.05" thick the system was showing a fairly high K_{eff} value of 0.940 ± .012 (i.e., Case H). The K_{eff} did not show a substantial drop in going from a cadmium thickness of 0.005" (Case G) to 0.05" (Case H) which is essentially a physical limit relative to mechanical dimensional constraints discussed in other sections.

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a. This condition was simulated by removing the water reflector in the system and, thus, preventing neutrons that had escaped from the container from incurring collisions that could cause reentry into the can fuel region.

b. Normally, this is taken at the 95% confidence level (i.e., 2σ).

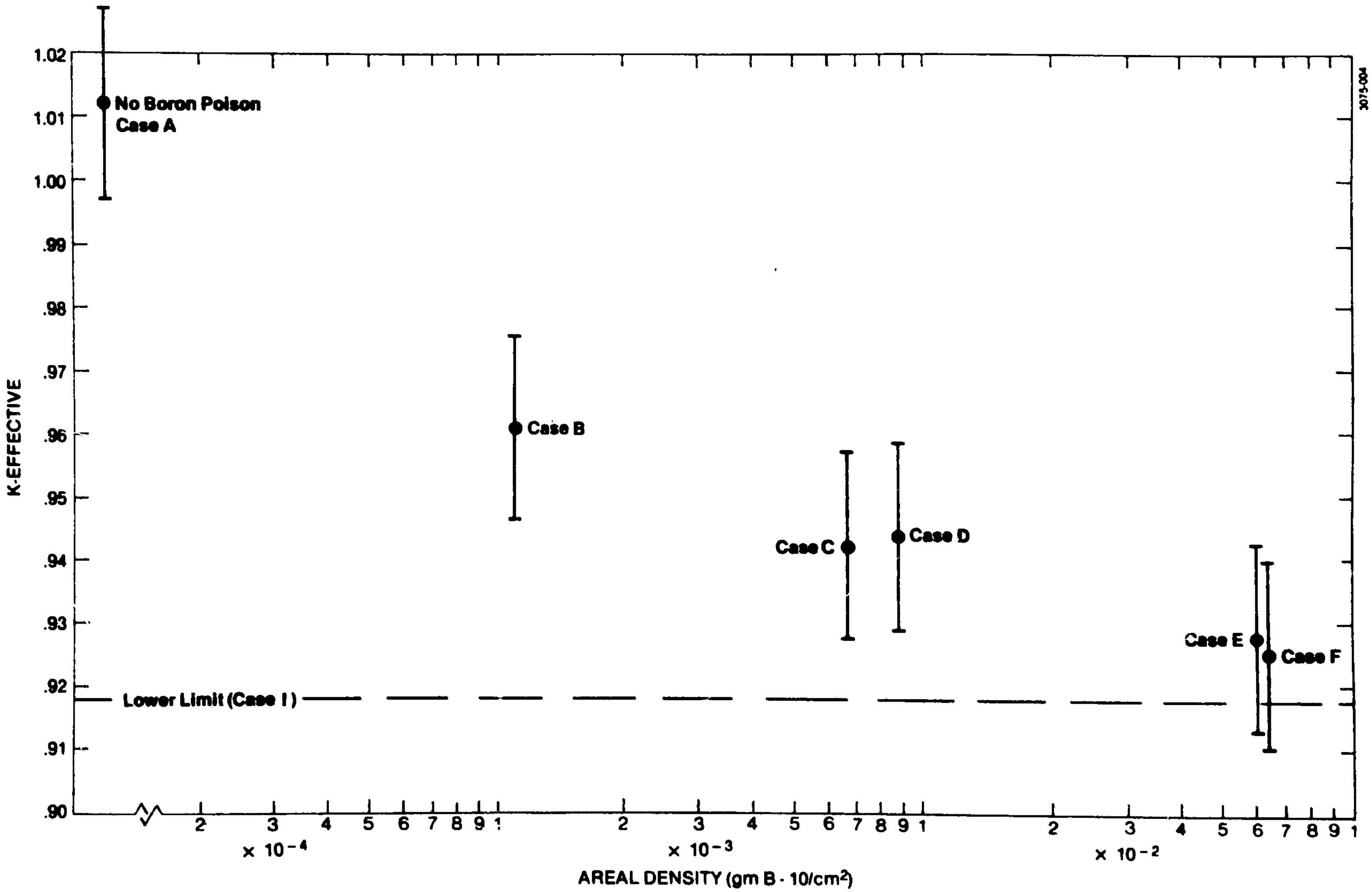

Figure 18. K_{eff} of a single canister versus the areal density (gram B-10/cm^2).

Effect of the Water-To-Fuel Pellet Ratio on the K_{eff}--All of the calculations reported in the Results - Single Canister in Water section above were performed with a fuel pellet-to-water volume ratio that is known to be the optimum (i.e., highest K_{eff}) for a system containing only pellets and water (i.e., no peripheral neutron poisons). Under these conditions, the fuel region of the canister was assumed to contain 84 grams U235/ℓ. Since boron and cadmium are basically thermal neutron absorbers, there is a possibility that a somewhat higher fuel pellet-to-water ratio than that used in the above calculations could lend its higher K_{eff} values of the poisoned canister system. To test this possibility, Case F was repeated with the exception that the fuel region contained a higher pellet-to-water ratio.[a] The calculated K_{eff} for the altered case was 0.846 ± .010, which may be compared to a value of 0.925 ± .012 for the standard case (Case F)..

In a detailed design "Final Criticality Assessment" of the poisoned round canister, more work is recommended to ensure that an optimum fuel pellet-to-water ratio has been established. However, based on a study of results from the calculation sighted above, no significant adverse effects are expected; and, the conclusions from the calculations reported in the Results - Single Canister in Water section above are expected to hold.

An Array of Canisters--The results in the Results - Single Canister in Water section above deal with a single round canister in water. Of course, for an array of round canisters, as in a storage environment, the neutron interaction between canisters will require consideration. A canister, square or round, will be essentially "isolated" from its neighbor provided that 8-10 inches of water is placed between units. On this basis, the K_{eff} of the array would be no higher than that of an individual unit.

To investigate the interaction effects, a calculation was performed using the conditions in Case E (i.e., areal density of 0.0608 grams B-10/cm^2) and assuming an infinite array of round canisters with two inches of water separating adjacent units. The calculated K_{eff} of the system was 0.956 ± .014. This value may be compared to the K_{eff} of 0.929 ± .014 for the individual unit (See Case E), which suggests an interaction effect of about +3% ΔK at the two inch spacing.

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a. The fuel density was raised from 84 grams U235/ℓ to 120 grams U235/ℓ.

It is noted that no such interaction effects calculation was run for the square canister analysis presented earlier. With the round canister case analyzed, the only effective parameter remaining for K_{eff} adjustments is spacing since the point of "diminishing returns" for poison has been reached making effective, additional poison use prohibitively costly. This may be noted graphically by reference to Figure 18 and imagining a smooth curve through the cases analyzed. In the case of the square canister, both poison and spacing are available for K_{eff} adjustments. Therefore, once a "licensable" K_{eff} is chosen for the canister storage racks, an optimization study between storage pool space "costs" and poisoned rack costs can be implemented which is typical of present high-density rack design technology. Such an optimization was felt to be beyond the present scope and more properly the cognizance of a rack vendor and GPU.

Conclusions--(1) Special neutron poisoning material (e.g., boron) will be required to meet individual round canister criticality safety objectives.

(2) Depending on the amount of poison included in the canister wall, the K_{eff} of a single canister will be in the range 1.01 to 0.92.

(3) An areal density of at least 0.007 grams B-10/cm^2 and, possibly, as high as 0.06 will be needed to achieve a K_{eff} in the vicinity of 0.95 for an individual canister in water.

(4) The criticality safety of an array of units, as in a storage environment, will require careful consideration for an areal density of 0.06 grams B-10/cm^2 and two inches separation distance, the calculated K_{eff} of the array is 0.956 ± .014. Since the K_{eff} of an individual unit will be relatively close to the acceptance limit of 0.95, there is not much margin for interaction effects and therefore greater than two inches of separation distance will be required.

Cask Shipping Criticality. The square and round canisters were examined for criticality safety in the shipping cask environment. The heavy metals associated with irradiated shipping casks (e.g., lead and depleted uranium are excellent neutron reflecting materials and can lead to higher reactivities than a water reflector.

For this examination the NLI shipping cask was employed. The model consisted of a single canister located in the center of the NLI cask. Aluminum was placed between the canister outside wall and the inside of the cask. The cask consists of concentric metal shells with dimensions as follows: 6 11/16 inches inside radius, 1/2 inch thick SS, 2 3/4 inches depleted uranium metal, 2 1/8 inches lead, and 7/8 inch SS.

Eight inches of water was placed on the outside of the container.

For the analysis, the computer code XSDRNPM[a] was used with the XSDRN 123 energy group neutron cross section set. All materials were assumed to be at 70°F.

Because XSDRNPM is a one-dimensional code, it was necessary in the case of the square canister to transform the square fuel assembly to a right cylinder by conserving the fuel volume. This geometric transformation slightly underestimates the neutron leakage (i.e., increase the K_{eff} slightly). The same fuel region conditions were assumed as in the other analyses reported above (i.e., fuel pellets in water at optimum spacing, no cladding).

For the square canister, the computed K_{eff} for the cask configuration was 0.99, which compares to a computed K_{eff} = 0.944 ± .011 for the case involving a water reflector (i.e., canister outside the cask and immersed in water).

For the evaluation of the round canister in the NLI cask, the boral poison associated with Case F (page 52) was employed (i.e., 0.155-inch thick boral located on the inside of the canister). The computed K_{eff} was 1.02, which may be compared to the value of 0.925 ± .012 for the poisoned canister with a water reflector.

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a. XSDRNPM is a one-dimensional, multi-group transport program that solves the Boltzmann transport equation by the method of discrete ordinates.

The relatively large K_{eff} difference between the canister in the cask versus the canister with a water reflector (i.e., 1.02 versus 0.925) is expected for the poisoned canister, since the boron poison is more effective in a water (i.e., thermal neutron) environment.

Alternative approaches are indicated and should involve candidate cask suppliers, GPU, and those involved in the next stage of canister design. This suggestion parallels that for the case of pool storage optimization studies discussed above.

Pressurization

Various mechanisms are possible that could potentially pressurize a sealed canister. They include release of volatile fission products, radiolytic decomposition of water, thermal expansion of gases, the solid "hydrostatic" internal pressure generated by UO_2 powder in a vertically oriented canister, conservatively assuming that a weakened intact fuel assembly containing all its plenum gasses fails after canister sealing, and external compression from lowering a sealed evacuated canister into an away-from-reactor (AFR) pool of 60-foot depth.

In Reference 1, it was shown that the heat load for a single TMI-2 fuel assembly with five years cooling (i.e., assumed earliest shipping date) was approximately 100 watts. In the NLI 1/2, LWT shipping cask this would be expected to generate a fuel temperature of approximately 110°F. If the canister is assumed to be filled with fuel debris, the heat load could be approximately 600 watts. Conservatively, using the fuel assembly model with a higher power level (i.e., assuming convection as primary transfer mechanism as opposed to conduction which would be the actual primary mechanism), the fuel debris temperature would be about 180°F. This would produce a pressure increase of the enclosed gases of less than 3 psi ΔP assuming the canister was filled at atmospheric pressure and room temperature prior to shipping. This produces stresses well below the 15,700 psi allowable for the material of construction which equates to approximately 14 psi ΔP for the square fuel canister design.

It was noted also that failure of all the fuel pins in an "intact" fuel assembly after canning would pressurize the square canister at approximately 14 psi ΔP, just at the allowable. Note, however, that "bulging" concerns in reference to dimensional interfaces such as racks, would only be temporary due to canister venting capabilities so long as elastic limits are not exceeded.

Two of the potential mechanisms for pressure buildup inside a failed fuel canister involve the release of volatile fission products and the radiolytic decomposition of water. Whereas, conditions have been identified (i.e., low temperatures and prior venting due to fuel cladding failure) which preclude any meaningful pressure contribution from the volatile fission products, water radiolysis could generate both hydrogen and oxygen. Unfortunately "G-values" for a sodium borate solution could not be located and probably do not exist. Any estimation using "G-values" from water or nitric acid solutions would be invalid and might lead to unsubstantiated conclusions. However, the combination of low-burnup, long-cooled fuel with internal temperatures less than 200°F would indicate little, if any, pressure increase during interim storage. Inclusion of the pressure relief mechanism in the canister design provides a safety feature for this potential. In addition, after having loaded the failed fuel into the canisters and providing for drainage, allowing additional drying to occur from self-heating before sealing should eliminate most of the retained water.

The same low temperatures inside the failed fuel canisters would be the reason for an insignificant release of volatile fission products from the uranium fuel matrix. After having already experienced temperatures in excess of 1000°C with probable total cladding failure, there is essentially no driving force for the release of any remaining volatile fission products retained by the fuel.

Calculations were also performed to determine the adequacy of the existing wall thickness for powdered UO_2 that could exert a solid "hydrostatic" pressure on the canister walls, particularly with the filled canister standing vertical. This indicated a requirement for approximately doubling the square canister 0.135-inch wall thickness. This could be provided by a sleeve. Internal crossties for the sleeve could be used to control bulging if required. Also, if the vacuum bag concept of Figure 8 was utilized, the necessary increase could be incorporated into the insert wall thickness.

Pressurization calculations for the round canister in accordance with the ASME Boiler and Pressure Vessel Code, Section VIII, Division 1, indicated internal pressure limits of 200 psi ΔP and external pressure limits of 50 psi ΔP for a wall thickness of 0.160 inches. These well exceed thermal gas pressure expected at approximately 3 psi as discussed above, solid "hydrostatic" internal pressure of approximately 66 psi, "intact" fuel pin rupture pressure of approximately 9 psi, and external compression force due to a 60-foot submersion at an AFR of approximately 26 psi.

Canister Sealing

Sealing of the cap to the canister for either cross section could be provided by using standard welding methods underwater implemented by a robot welder within a welding bell. The concept is illustrated in Figure 19. The canister itself indexes the travel of the welding fixture during sealing. Prior to welding, inert gas lines attached to the bell would displace the water from under the bell and provide the inert atmosphere required for stainless steel welding. It is not anticipated that potential borated residues or minor quantities of trapped water in the vicinity of the welding would significantly impact the finished weld which would be required only to provide sealing for pressures of 50 to 100 psi. The initial heat should remove most trapped water. However, weep holes could be provided to drain trapped water in the welding joint if more desirable. Multiple passes (using fillers if warranted) would ensure hermetic seals. As an alternate to the bell, an evacuated fixed chamber could also be used.

Sealing of the vent, purge, and sampling point could be done with threaded plugs and luting compounds.[16]

Other sealing methods are possible but could require more development than adopting standard methods to the underwater environment required in this application. Included would be explosive seam welding.[17,18] This has been implemented on thin wall stainless steel components comparable to the canister

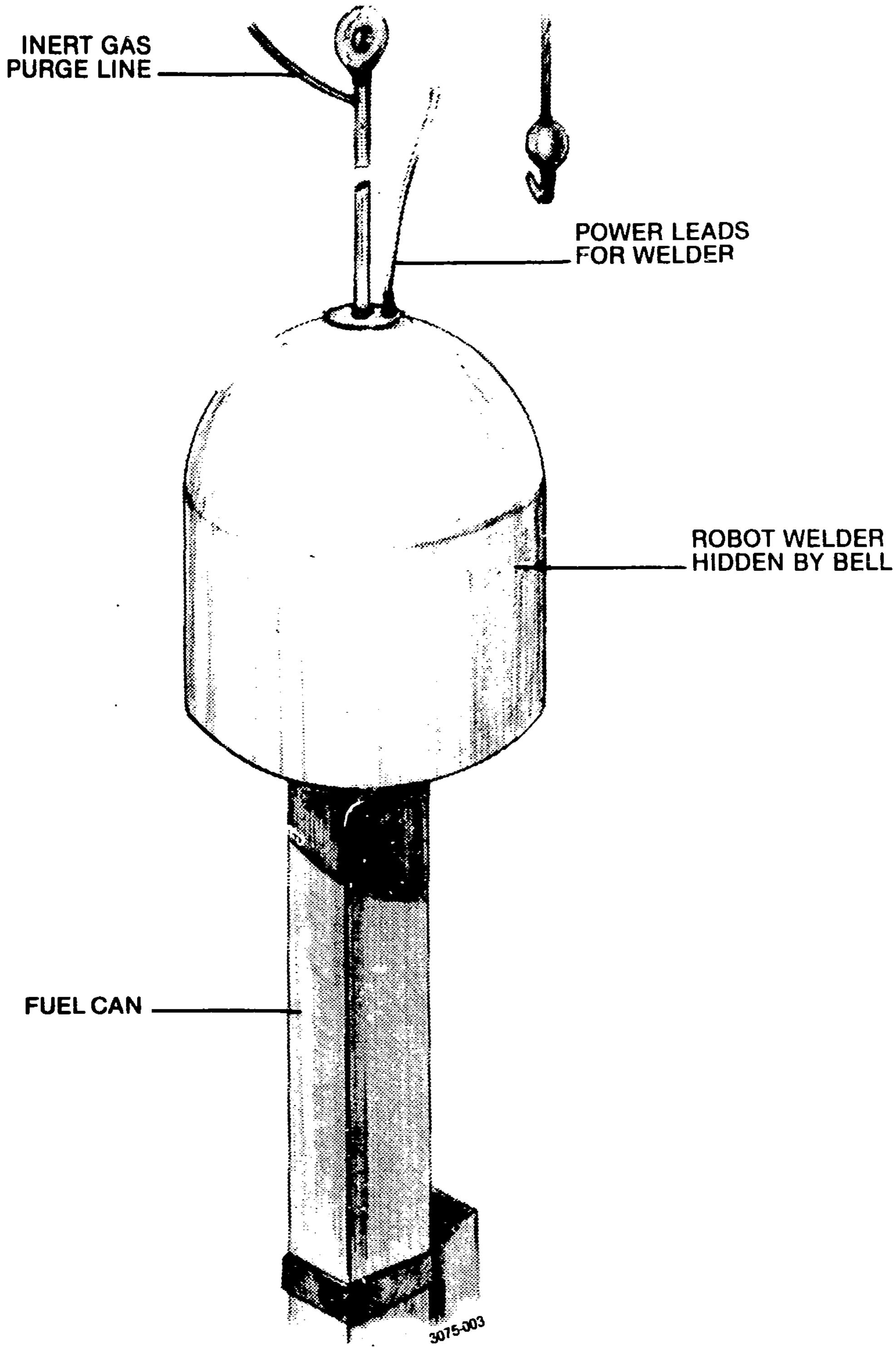

Figure 19. Welding bell.

wall thickness of these alternatives and cap-to-canister machined contact surface lengths comparable to these alternatives also. Explosive ribbons using very small amounts of explosive (10 to 20 grains/foot) are remotely detonated joining machined interfaces. Subsequent helium leaking testing has confirmed hermetic seals within the pressure ranges required for this application.

Material

Based on the following corrosion discussions which indicate no overriding justification to select titanium or Zircaloy, stainless steel was then selected as the canister material. This selection is primarily based on stainless steel being an order of magnitude less expensive than these more exotic materials. Since these same discussions indicate that crevice corrosion and pitting are unlikely, the selection of 304L or 316L appear adequate and essentially equal from a cost and availability point of view. The only cost competitive candidate to these was aluminum which has shown corrosion in borated pools.[19]

It is also concluded that if the wet storage temperature is held to less than 200°F, which is, of course, most probable, the fuel canister welds will not need to be stress relieved. This is a conservative, qualitative figure; the actual limiting temperature is probably greater.

Borated Water. The temporary storage of the fuel, perhaps one to five years, would most likely be on-site and within the existing TMI Fuel Storage Pool. This pool normally has the following water chemistry:

pH range	5.2-5.5
Cl^-, ppm	0.01
Boron, ppm	2120-2140
Lithium, ppm	low (n.a.)

The pH is a bit lower than most other domestic fuel storage pools, but the most notable difference is the higher content of boron in the TMI pool compared to other pools. The boron has little effect on stainless steels, titanium, or zirconium alloys, but aluminum corrosion has occurred in some borated pools.[19]

An increase in pH from the normal case of 5.2-5.5 to the current TMI pool pH of 8.0 decreases the possibility of stress corrosion cracking but increases the possibility of pitting. The increased boron concentration (3500 ppm from about 2130 ppm) would have little, if any, effect upon increasing corrosion failure. Sodium hydroxide involved in these pH changes should not, of itself, cause problems. However, the level of chlorides present as impurities must be controlled as discussed below.

Crevice Corrosion and Pitting. Crevice corrosion frequently occurs within crevices and other shielded areas on metal surfaces exposed to corrosives. This type of attack is usually associated with small volumes of stagnant solution caused by holes, gasket surfaces, bolted joints, surface deposits, and crevices under bolt and rivet heads. Stainless steels are particularly susceptible to crevice attack. However, in the case of the TMI fuel canister, crevice corrosion is very unlikely. We do not have stagnant pool conditions, the pool water is not highly corrosive, and the crevices and shielded areas are designed out of the fuel canister.

Pitting is a form of extremely localized attack that results in holes in the metal. In the case of stainless steel, the chloride ion is the substance that usually causes pitting. Two other parameters that affect the rate of pitting attack are temperature and the pH of the solution in which the metal is placed. As temperature and pH increases, the rate of pitting increases. Pitting is usually associated with stagnant conditions also. Chloride exposure cannot be ruled out and the pH in the TMI pool is slightly higher than normal. On the positive side, the pool is relatively cool and not stagnant.

In summary, crevice corrosion is unlikely and pitting is possible, though still unlikely.

Stress Corrosion Cracking. Sensitized types 304 and 316 stainless steel developed stress corrosion cracking in borated solutions (3000 ppm B) at pH's of 4.5 to 7.5 and chlorides in the range of 5 to 200 ppm at temperatures of 80 to 140°C.[20] The normal chlorides in the TMI pool, which are only 0.01 ppm, are probably too low to present any problem. Stress corrosion cracking could be a problem due to the relatively high levels of boron, about 3500 ppm, but one must keep in mind that the phenomena of stress corrosion cracking is also dependent upon the stress and temperature involved. In the case of a fuel canister in a storage pool, there will be little stress on the canister (only residual welding stresses) and the canister will be stored at relatively low temperature. The use of low-carbon stainless steel, such as 304L, and stabilized alloys decrease the prospects that stress corrosion cracking will occur during pool storage.

Galvanic Couples. Experiments with about 50 different metal galvanic couples in boric acid solutions at 50 to 300°C were referenced.[21] The only couples which showed evidence of galvanic corrosion were those involving aluminum, 4340 carbon steel, boronated stainless steel, boral, and nickel-plated 80 Ag-15 In-5 Cd. In general, corrosion and boron absorption experiments have shown carbon steel and aluminum to be unacceptable materials for most nuclear industry applications.

The generally passive nature of the stainless steels permits coupling to other passive materials such as Inconel and Zircaloy without significant corrosion due to the couple.

Hydriding Effects in Zircaloy. Zirconium alloys form hydrides which are brittle and provide a means for cracks to propagate. The phenomenon appears to require high stress intensity factors. At 75°C, the threshold stress for cracking is approximatey 80,000 psi for Zr-2.5 NB. The crack propagation rate increases with increasing temperatures.

A fuel canister made of Zircaloy would quite likely experience no hydriding and subsequent crack propagation because of the low storage temperature and stresses involved. This is particularly so if the fuel can has had the weld (assuming weld closure) stress relieved.

Hydriding Effects in Stainless Steel. The hydriding of stainless steel appears to be even less a problem than for Zircaloy. Hydrogen solubilities for stainless steel are low even though hydrogen permeation is relatively high at reactor operating temperatures.

If the fuel canister were to be fabricated of stainless steel, there does not appear to be a hydriding problem, either from the fuel or the pool environment. Many austenitic stainless steels (e.g., 304) tend to resist hydrogen cracking.

Fission Product Attack. Laboratory studies have shown iodine and cesium to cause cracking in Zircaloy. The fission product attack requires high stress levels and high temperatures such as occur in the fuel rod during irradiation. The fuel canister that contains the TMI fuel is not expected to be attacked by fission products due to the relatively low temperatures to which the canisters will be exposed.

Cost

A cost analysis was made of the design shown on Figure 15. Based on the following, a cost per unit of $2000 is estimated without debris inserts.

(1) $2.50 pound for the 304L SS material
(2) $25 per hour labor
(3) 10% QA/QC
(4) 10% profit
(5) 250 unit order with tooling setups included.

The details of this cost estimate are presented in Appendix A. Assuming 40 debris inserts and spreading their cost over the total order, the unit cost would be approximately $2200.

The round canister cost estimate is based on revisions to the square canister detail estimate. The above assumptions hold except a lesser quantity is anticipated. It is further assumed that the provision of the cap mechanical attachment system and the drain systems will equal those costs for the square canister. The revision is then based on an increase in the material required for each round canister unit and a decrease in the labor required since only one lengthwise seam weld is now needed for the two required on the square canister. The result is a unit cost decrease of approximately $350 per unit. The details of this cost estimate are also presented in Appendix A. However, based on the criticality calculations performed herein, to the resulting $1650 must be added the poison sleeve costs noted in Table 2 to arrive at a total round canister unit cost which would not be less than $3000.

Interface Constraints

The design developments per Figures 15 and 16 attempt to minimize the constraints on equipment and systems with which they must interface. These include tooling for handling empty and loaded canisters, tooling that will load the canisters, racks that store the canisters, accountability and transportation systems and final disposition systems. The round canister alternative, for example, should facilitate loading relative to the square canister. However, the square canister does not require poison for water reflected environments and would be more easily handled at a reprocessing facility.

Canister Handling

Equipment and systems must be developed in accordance with the dimensional constraints of Figure 15 for a loaded square canister design weight of 6000 pounds or in accordance with the dimensional constraints of Figure 16 for a loaded round canister design weight of 8500 pounds as follows:

(1) Empty canister grapple

(2) Cap grapple or hook

(3) Canister holder(s)

(4) Compatible intact-plus-non-fuel-bearing-component fuel loading tooling (e.g., see Single Multi-Application Canister section above).

(5) Compatible large or sectional debris loading tooling

(6) Compatible large and fine debris hydraulic suction loading tooling

(7) Powdered debris wall reinforcement inserts for the square canister only

(8) Cap positioner

(9) Cap/canister welding system

(10) Purge system and connectors

(11) Sample system and connectors

(12) Purge and sample plugging tooling

(13) 3- or 5-Ton crane handling system.

Various potential mechanisms for these purposes have been discussed already in the Single Multi-Application Canister section above. Additional intact fuel grappling equipment not specifically mentioned therein could be envisioned. Details that must be compatible with the canister development may include fuel grappled via the lower portions of clear thimble tubes for vertical loading or fuel grappled via the lower spacer grids by a comb equal in depth to the fuel width or fuel grappled via the end fittings.

Defueling Equipment

An evaluation of the interfaces between the developed fuel canister and the fuel transfer carriage basket was performed by reference to Babcock and Wilcox drawings of the fuel transfer system that included: 44-54-009-03, 44-54-011-02, 44-54-017-08, and 44-54-040-30.

The evaluation was used to develop any limiting or potential dimensional interfaces such as the maximum canister length or largest allowable cross-section.

The maximum packaged fuel length that could be accommodated by the carriage basket was investigated. Assuming the stop assembly (Item 1170, 44-54-011-02) could be relocated, the limiting interference would be the flange of the transfer tube (Item 3367, 44-54-040-30) on the reactor side. Use of this maximum length might require removal of the pool emergency cable system (Item 0022, 44-54-017-08), but insufficient detail was available to say with certainty. The maximum, uninterfered length would be 15 feet 1/8 inch. The maximum square cross-section that could be contained within the 15.25-inch octagonal inside diameter of the basket would be 11.48 inches. Therefore, the square or round fuel canisters could be accommodated "as are." However, the structural adequacy of the carriage for this 6000- or 8500-pound load would require verification.

Suggestion has also been made for a "box" carriage that does not rotate as the present upender.[22] This influenced the vertical/ horizontal "tilting" capability of the canisters. Any required structural increases might be incorporated in the "box" carriage modification.

Fuel Storage Racks

Specially designed packaged fuel racks will be required for either fuel canister. (Burns and Roe Drawing W.O. 2555/2066 R.15). Typically, for seismic reasons, racks are limited to 3/8 to 1/2 inch maximum cross-sectional clearance between the fuel and the rack. In this case, the square fuel canister would not fit existing slots in the Burns and Roe design. Obviously, the round canisters would require new racks. Therefore, new racks must be designed to accommodate both of these canister alternatives using state-of-the-art technology for high density storage. Design, licensing, and fabrication could take 18 months or more. Projections for AFR rack licensing have indicated periods of 3 years or more as possibilities.

Accountability

Probably the biggest factor which affects the material control and accounting requirements is the quantity of debris which may have developed as a result of fuel element damage. This uncertainty, therefore, imposes a need to provide for a reasonably attainable method to establish an approximate quantity of debris. Because various adverse conditions exist including loss of identity and significant fuel as debris, little if any NDA verification within accuracy limits suitable for accountability purposes is possible. Since adequate visual observation may be obstructed, a relatively accurate method of weight loss should be provided to indicate the extent of change to each assembly.

All canisters, therefore, should be tare weighed dry and underwater. The weight of the loaded canister should also be obtained directly after loading and after dewatering. This, of course, includes debris canisters.

Provisions for observation with television cameras, video-taping, and logging may be required. The fuel assembly identity should be verified, if possible, and displayed along with the canister identity so that both could be observed at the same time. Similar conditions could apply to debris collection activities also. Provisions for application of tamper-safe seals on the canisters have been made. These include nonstandard socket plugs. In the case of the square canister, the cap is mechanically locked once it is seated on the canister. Crimping of the round canister cap and side wall could provide similar tamper-safe approaches for the round canister.

Canisters are identified on top and bottom and on all four sides for square canisters or at 120° for round canisters near the top with permanently engraved one-inch high digits which will not corrode after long-term underwater storage. Weld seams should be minimized by the selected fabrication technique to allow more obvious detection of any repaired diversion attempts.

Of course, many of the above features are responses to assumed requirements. Fixed requirements are contingent on third party concurrence which is difficult to assess with certainty.

Nondestructive Assay

The design, configuration, and materials of construction of the unpoisoned square spent fuel canister would have only secondary effects on the accuracy and sensitivity of NDA methods if used. However, the use of a poison in a spent fuel canister could adversely affect nondestructive methods involving passive or active neutron assay techniques.

The expected uncertainty associated with NDA measurements of canned TMI-2 fuel using current technology is estimated at approximately 10-15 percent. Measurement of the cesium-137 activity from both LWR and BWR irradiated spent fuel assemblies has been performed experimentally and results compared with declared burnup values. The average percent agreement was found to be ±5% at the 95% confidence level. Since these measurements were performed under better conditions than would be expected for the TMI-2 damaged fuel, similar uncertainties should not be anticipated.

Complicating features which tend to adversely affect the sensitivity and accuracy of the measurements include: (1) the presence of the soluble poison boron, whether in a liquid or as a solid; (2) gamma-ray attenuation; (3) low burnup and long cooling period; and (4) inclusion of the burnable poison rods. Additional problems introduced by the necessity for encapsulation of the core in debris form include the preferential leaching of specific fission products from the fuel matrix thereby destroying a fixed fission product-fuel material ratio and the nonuniform attenuation of gamma rays due to random localization of nonfuel bearing components. Because of these complications and measurement uncertainties, NDA measurement of encapsulated core debris does not appear promising as an accountability method.

Further Disposition

A generic interface evaluation of the packaged fuel with an off-site receiving facility was made. The receiving facility could be assumed to provide some of the following functions for the fuel:

(1) Analysis of Fuel Condition
(2) Fuel Storage
(3) Chemical Reprocessing
(4) Ultimate Disposal as Waste.

The facility must be able to receive and unload the shipping cask. As a result, the design of the fuel canister should be small enough to minimize hindrance with implementation of the above four functions and to permit the cask receipt to be made. The following minimal canister criteria were established. The canister system should:

- Provide pressure retaining capability and contamination control

- Provide internal sampling capability to the canister to monitor gas inventory (e.g., moisture content, hydrogen concentration, decay gas composition, etc.)

- Permit a dewatering of a wet canister and alternately a water filling capability via a fill and drain system.

A single multi-application canister (whether square or round) with a welded end cap providing vent, fill, and drain capability for the enclosed fuel would meet the first three requirements. It would not accommodate these possible requirements:

- To provide for simple remote mechanical wet or dry unpackaging

- To provide alternative canister cross-sections to preclude recanning for more efficient storage, to accommodate handling preparatory to reprocessing, or to prepare for hot cell analysis and examinations.

Due to potential space and equipment limitations potential receiving facilities might more easily mechanically uncap and/or unpackage the fuel using canisters that open or disassemble mechanically. For example, a research facility that must unpackage the fuel may have more difficulty providing remote, metal-cutting equipment suitable for a canister of this size that could require full length cutting. Also, it is probably more likely that the facility lacks the room or has insufficient size (typically a length equal to twice the canister length is needed to permit axial uncanning). Here, a specially designed canister that parted on an axial centerline could be envisioned. However, this design would compromise the degree of containment integrity provided by the welded cap on the developed canisters and based on current information appears unwarranted.

Reprocessing and storage facilities may not be able to handle canisters with the square cross section dimensioned as in Figure 15 or round canisters dimensioned as in Figure 16. However, for storage facilities complete interface evaluations would be required on a site specific basis. Interface items to be addressed for these proposed facilities would include among other things:

- Dimensional suitability
- Structural and seismic integrity
- Criticality evaluations for storage arrays
- Thermal environment
- Corrosion environment.

Since facilities selection is not feasible at this stage of the study, the first three requirements for the canister will be assumed controlling. In any case, if additional handling flexibility is desired in the canning system, this must be decided before the design of the package is finalized.

Square Versus Round Canisters Selection

First a round canister option from Table 2 must be selected with Case IV appearing to have the best blended features. Then Table 3 may be referenced for a qualitative listing of considerations between these two single multi-application canister options. In the final analysis, licensability must be weighed against canister loading. Neither is automatically eliminated on either point nor on secondary considerations. The selection is properly the perogative of the party responsible for implementing both of the above major functions to a successful completion.

TABLE 3. ROUND VERSUS SQUARE CANISTER TRADE-OFF CONSIDERATIONS

	Round	Square
Licensability	Harder	Easier
Criticality Control (water reflected)	Neutron Poison	Geometry
Can Loading	Easier	Harder
Cost/Unit	>35% More	Less
Number of Units	Less	More
Capacity	Larger	Smaller
Max. Wt/Unit	8500 lbs.	6000 lbs.
Temporary Mechanical Seal	Twist With Gasket	Potential "Roll Out" With Gasket
Seal Back-Ups	"Below Boiling" Fusible Insert	Same
Seal Welding	Easier	Harder
Pressurization Limits	Higher	Lower
Side Drain Interference	More Likely	Less Likely
Racking	Requires Rack Development and Licensing	Requires Rack Development and Licensing
Cask Dimensional Compatibility	Potentially Limited (i.e., Case V Table 2)	Unlimited
Cask Criticality Control Compatibility	Not Defined	Not Defined
Spent Fuel Pool Space Requirements	Not Defined	Not Defined
Capping	Rotation	Linear Push

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APPENDIX A

TMI FAILED FUEL

CANISTER PRELIMINARY COST ESTIMATES

SQUARE CANISTER PRELIMINARY COST ESTIMATE

SUMMARY

Item	Quantity	Material (A)	Labor Hours (B)	Tooling (C)	Total (A)(B)(C)
1	500 T&B PL	18375	845 x 25 = 21125	500	40,000
2	250 Bail	1400	65 x 25 = 1625	200	3,225
3	250 Sample Pt	300	151 x 25 = 3775	1100	5,175
4	250 Press.Relief	19150	405 x 25 = 10125	1600	30,875
5	250 Can	30750	7725 x 25 = 193125	1600	225,475
6	250 Re-Strap	848	1850 x 25 = 46250	1400	48,498
7	250 Cap Bands	1525	1459 x 25 = 36475	1900	39,900
8	250 Misc Weld	--	792 x 25 = 19800	--	19,800
				TOTAL:	412,948

250 ⟌ 412,948 = $1651/can

x 1.10 QA/AC

$1816/can

x 1.10 Profit

$1997.6

This cost estimate assumes the following:

(1) Material Cost of $2.50 per pound.

(2) Labor Cost of $25 per hour.

(3) 10% QA/QC.

(4) 10% Profit Markup.

(5) Production of 250 cans.

1. Top and Bottom Plate 9.16 x 9.16 x 1/2" tk

500 pieces

Fab Process:

- Cut blanks from stock

Top (250)	Bottom (250)
Bevel sides $3\frac{1}{2}$ ft/blank	Bore 1 hole
Bore 2 holes	

Material: $9\frac{1}{3}$ x $9\frac{1}{3}$ = 297 ft^2

Stock: 3(at 72" x 240")sheets

Assume \$2.50/lb - \$6125/sheet = \$18375/3 sheets

Labor:

- Cut Blanks ($\frac{1}{2}$ hr/blank)(500) 250 hrs
- Bevel Edges ($\frac{1}{4}$ hr/ft)(3.5)(250) 220 hrs
- Bore Holes (10 min/hole)(750 holes) 125 hrs
- Tap NPT (15 min/hole) (250 holes) 250 hrs

TOTAL: 845 hrs

Tooling:

Consumables \$500

2. Bail 12"L x 3/4" Ø

12" x 250 Say: 275 L.ft. at \$5.09/ft (McMaster-Carr) = \$1400

Material: \$1400

Fab Process:

- Cut to length
- Bend to shape

Labor:

- Cut (10 min. to cut)(260 cuts) = 43 hrs
- Bend (5 min. to bend)(260 bends) = 22 hrs

TOTAL: 65 hrs

Consumables $200

3. Sample Port and Filter 1-3/4"Ø x 2"lg (McMaster-Carr)

Material: 2" lg x 250 = 10 ft. $303/12 ft, say: $303

Fab Process:

- Cut length
- Bore 5/8" hole
- Counterbore 1½" hole
- Tap for 3/4" NPT

Labor:

- Cut (10 mi˞,cut)(260 cuts) = 43 hrs
- Bore 5/8" Ø (10 min/part)(260) = 43 hrs
- Counterbore (15 min/part)(260) = 22 hrs
- Tap to 3/4"Ø (10 min/part)(260) = 43 hrs

TOTAL: 151 hrs

Tooling:

- Jig for drill presses $600
- Consumables $500

TOTAL: $1100

4. Pressure Relief Valve Assembly

Components: Valve \$75 ea x 250 = \$18750
(McMaster-Carr)

2" Ø Pipe Sch. 10 x 2" L = 10 ft \$120 \$120

2.35" Ø plug plate 11 ft^2 at ¼" tk Say: \$280

$\left(11\ ft^2 \times \frac{.25''}{12''/ft} \times 490\ \#/ft^3 \times \$2.50/lb\right)$ \$19150

Fab Process:

- Cut and Bevel (20 min)(250) = 83 hrs
- Cut Strip Stock (15 min/ft)(50 ft) = 12.5 hrs
- Stamp Top (4 min)(250) = 16.6 hrs
- Tap Top (10 min)(250) = 43 hrs
- Weld 2' Bevel (1 hr)(250) = 250 hrs

405 hrs

Tooling:

- Stamp and Die \$1000
- Consumables \$ 600

\$1600

5. Can 9.430 " □ x 173½ 14.45 ft. 8.923 ft^2 10 ga.

49.2 lb x \$2.50/lb = \$123

\$123 x 250 = \$30750

Fab Process:

- Break 2 L's
- Stamp Retaining Slots
- Weld 2 L's - seam

Labor for Break

- (30 min/Break)(500) = 250 hrs
- Stamp Retaining Slots (30 min per L)-500 = 250 hrs
- Weld and Weld Insp. (1 hr/ft)
  (1 hr/ft)(28.9)(250) = 7225 hrs
  (Also incl. corner & radus welds)

7725 hrs

Tooling Jigs for Break: $400

Stamp and Die: $1200

$1600

6. Reinforcing Strap

$9.55" \times 4 \text{ sides} \times 2" = \frac{76.44 \text{ in}^2}{144 \text{ in}^2/\text{ft}^2} \times 250 = 132.7 \text{ ft.}$

$132.7 \text{ ft}^2 \times \frac{.0625}{12} = .69 \text{ ft}^3$

339 lb x 2.50/lb = $848

Fab Process:

- Break 2 L's
- Weld 2 L's
- Stamp 38" x 2" x .0625" Strips

Labor:

- Break (15 min/break)(500) = 125 hrs
- Stamp Mat'l (15 min/L)(500) = 125 hrs
- Weld & Weld Insp. (1 hr/ft)
  (1 hr/ft)(6.4 ft)(250) = 1600 hrs

1850 hrs

Tooling Jigs for Break: $400

Stamp and Die $1000

$1400

7. Cap - 1/8" Band and Clip

Band: 1-5/8" x 9.5 x 4 x .135 x 250 =

1.206 ft^3 x 490 lb/ft^3 x $2.50/lb. = $1477

Clip: .09" x 3/8" x 2" x 4 sides x 250 caps =

19 lb x 2.50/lb = $48

TOTAL: $1525

Fab Process:

- Break 2 L's
- Weld 2 L's
- Weld sq to 1/2" R
- Machine Clips
- Weld Clips to Band

Labor:

- Break (15 min/break)(500) = 125 hrs
- Stamp Mat'l (15 min/L)(500) = 125 hrs
- Machine Clips (15 min/clip)(250 x 4) = 250 hrs
- Weld Clips (1 hr/ft x .666 ft x 250) = 167 hrs
- Weld Band to 1/2" PL 3.16 x 250 = 792 hrs

TOTAL: 1459 hrs

Jigs for Break: $400

Stamp and Die: $1000

Consumables: & Misc. $500

$1900

8. Weld Bottom PL (1/2")

3.16 ft x 1 hr/ft x 250 = 792 hrs

DEBRIS INSERT ESTIMATE

Assume 40 units

Reference I. 5 above.

Material:	Same as in Item 5 ($123) (40 units) = $4,920
Fab Process:	Delete Stamping
Labor:	Breaking (1/2 hr/break) (80) = 40 hrs
	Welding (1 hr/ft) (28.9) (40) = 1,156 hrs
	1,196 hrs

Tooling: No cost
Total Cost: $4,920 + (1,196) (25) = $34,820
% Increase: ($34,820) ÷ ($412,948) = 8.4%
Unit Cost Increase: (8.4%) ($1,997.60) = $168

ROUND CANISTER PRELIMINARY COST ESTIMATE

Assume square can cost details except as noted below:

1. Caps and drain system costs equal though different in mechanical detail.
2. Costing differences limited to can body material quantities differences and elimination of one lengthwise seam weld and various fab steps.
3. Breaking costs equal rolling costs.
4. Can thickness based on borated stainless steel unit, not minimal structure thickness.

Material:

5% circumferential increase

80% thickness increase

Reference Item 5. above: Material Cost = $30750

Cost Increase = ($30750)(0.85) = $26,138

Unit Cost Increase = ($26,138) ÷ (250) = $105

Labor and Tooling:

Delete stamping and 1 seam weld

Cost Decrease = (250 hrs + 3613 hrs)($25/hr) + $1200 = $97,775

Unit Cost Decrease = ($97,775) ÷ (250) = $391

Total Unit Cost Decrease = $391 - $105 = $286

Total Unit Cost Decrease Factored by QA/QC and Profits = ($286)(1.1)(1.1) = $346

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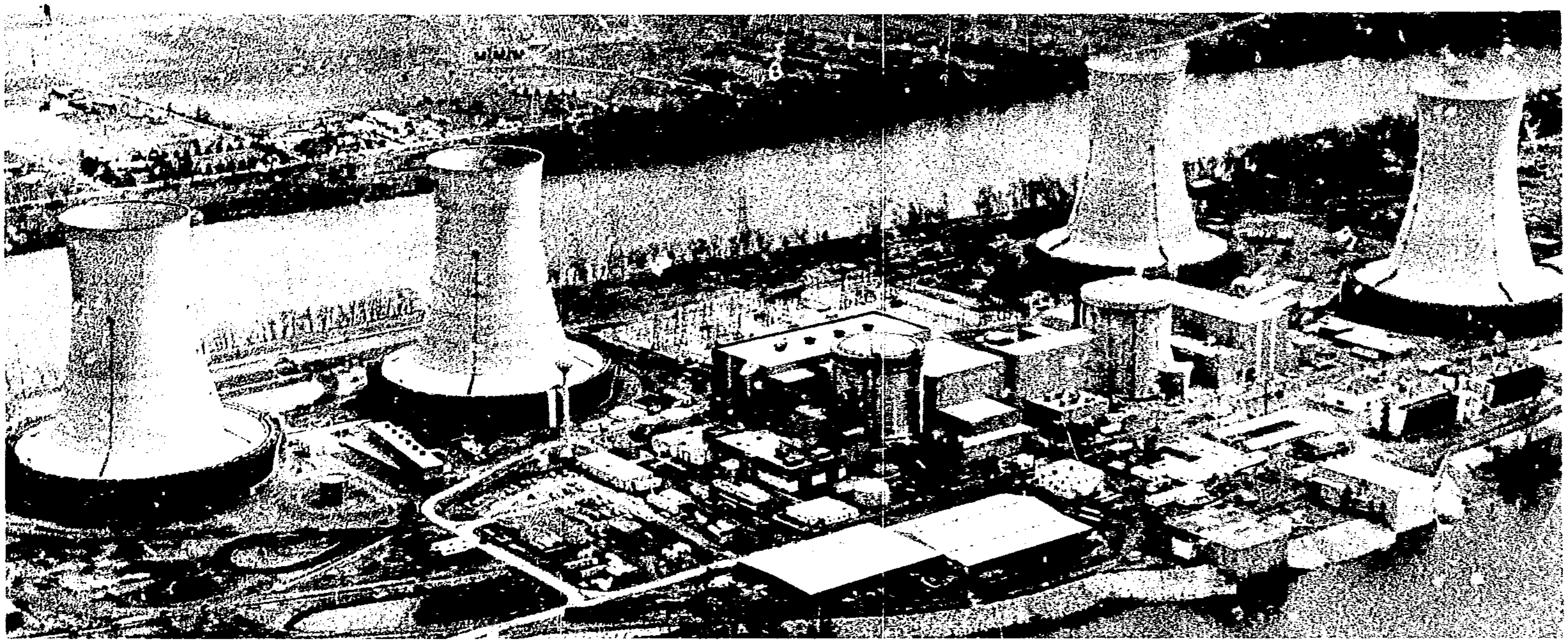

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General Public Utilities • Electric Power Research Institute • U.S. Nuclear Regulatory Commission • U.S. Department of Energy

TMI-2 Reactor Building Purge—Kr-85 Venting

Lawrence J. Kripps

March 1981

Prepared for EG&G Idaho, Inc.
Under Subcontract No. K-5154
and the U.S. Department of Energy
Under Contract No. DE-AC07-76IDO1570

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ABSTRACT

A comprehensive technical report of the total effort involved in the decontamination of the Three Mile Island Unit 2 (TMI-2) Reactor Building atmosphere by venting the contained Kr-85 to the environment is presented. This technical documentation is intended for inclusion in the Technical Integration Office (TIO) data bank as a TMI-2 on-site cleanup activity of interest to the Information and Examination Program. The scope includes the licensing effort which was required to obtain NRC's approval to vent, a description of the plant equipment and instrumentation involved in the venting operation, how the venting was controlled to conform with technical specifications, problems encountered during venting, a summary and analysis of pertinent venting data, and a description and results of the on-site and extensive off-site radiological environmental monitoring programs conducted during the Kr-85 venting.

ACKNOWLEDGEMENTS

This report documenting the TMI-2 Reactor Building atmosphere purge would not have been possible without the cooperation and support received from Mike Morrell, the Met-Ed/GPU special projects engineer in charge of the Reactor Building purge program. Throughout the two months the author spent on-site at TMI before, during, and after the venting, Mike Morrell was a constant source of information, and later he provided several critical reviews of this report to ensure its completeness and accuracy. Additionally, thanks are deserved by Jackie Tate of Gilbert Associates, who served as a purge shift engineer and always seemed available to answer the author's questions, by Henry Peterson, of TMI's Radiological Technical Support group who supplied information on the effluent sampling program and on-site and Auxiliary Building radiological monitoring programs, and by Bill Riethle and Gary Baker of the TMI Environmental Controls Group who supplied the description and results of the Met-Ed/GPU Radiological Environmental Monitoring Program during the venting.

The author would also like to thank all the people from EG&G, the EPA, the Pennsylvania Department of Environmental Resources, NRC, Met-Ed/GPU, and technical consulting companies to Met-Ed/GPU who were involved in the Reactor Building purge and who were so helpful to this author as he tried to document the overall venting effort. And a special final thanks is due to Judy Lancaster, the author's secretary who spent many hours typing and retyping this report.

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SUMMARY

Between June 28 and July 11, 1980, aproximately 44,000 curies of Kr-85 were released to the environment during a controlled purge of the TMI-2 Reactor Building. The removal of Kr-85 from the Reactor Building atmosphere was a necessary precursor to the ultimate total decontamination of the TMI-2 facility. It also removed the risk of unpredictable and uncontrollable Kr-85 releases and allowed less restricted Reactor Building access. Following the decision by Met-Ed/GPU that controlled venting of the Reactor Building was the safest alternative for removal of the Kr-85, it required over seven months of intensive licensing effort to finally receive NRC's approval to commence venting. The licensing process included the preparation by NRC of an Environmental Assessment of the proposed venting plan and involved a significant amount of public participation.

The purging of the Reactor Building was accomplished using two existing systems that only required slight modifications. The Modified Hydrogen Control System and the Modified "B" Train of the Reactor Building Air Purge and Purification System were used to vent the Reactor Building at a slow or fast rate, respectively, depending on the Reactor Building Kr-85 concentration and existing meteorology. The venting flow rate was controlled so as not to exceed an off-site integrated dose of 15 mrem beta skin or 5 mrem total body or an off-site dose rate more than 3 mrem/hr beta skin or 1 mrem/hr whole body. All releases were through the station vent which contained instrumentation to accurately monitor the release of all radioactive materials. Except for some initial effluent particulate monitoring problems resulting from a high Kr-85 background, the entire venting operation went smoothly.

The final results of the effluent monitoring showed that of an original 44,600 curies of Kr-85 contained in the Reactor Building (range 43,000 to 46,200 curies), 44,132 curies of Kr-85 were vented (range 38,302 to 50,254 curies). Also an estimated 1.3 curies of tritium, 5.5 E-6 curies of Cs-137, and 5.72 E-9 curies of Sr-90 were released.

A large radiological environmental monitoring effort conducted by Met-Ed/GPU, EPA, and others all confirmed that the detectable off-site releases of radioactive

material and their resulting doses were well within the Technical Specifications set by the NRC. Met-Ed/GPU also ensured through an on-site and Auxiliary Building monitoring program that personnel on site were not exposed to radiation in excess of permitted dose limits.

Although not within the scope of this report it should be noted that following the main Reactor Building purge a number of smaller subsequent purges occurred. These mini-purges were required because the Reactor Building Kr-85 concentration increased by approximately a factor of 100 following the June 28th to July 11th purge. The reason for this is thought to be Kr-85 coming out of the sump water.

1.0 INTRODUCTION

As a result of the March 28, 1979 accident at Three Mile Island Unit 2 (TMI-2), significant quantities of radioactive fission products, including gases, particulates, and iodine, were released into the enclosed Reactor Building atmosphere from failed fuel in the reactor core. The airborne radioactivity within the Reactor Building gradually decreased following the accident because of the decay of the short-lived radioactive fission products such as xenon and iodine. The principal remaining radionuclide in the Reactor Building atmosphere before its decontamination was Krypton-85 (Kr-85) which has a 10.7 year half-life.

To permit the less restricted access to the Reactor Building necessary to gather information, to maintain instrumentation and equipment, and to proceed toward the total decontamination of the TMI-2 facility, Metropolitan Edison/General Public Utilities (Met-Ed/GPU) on November 13, 1979 asked NRC for permission to remove the Kr-85 contained in the Reactor Building atmosphere by venting it to the environment. Met-Ed/GPU supported their request with a Safety Analysis and Environmental Report of the proposed Reactor Building venting plan. After seven months of protracted licensing efforts which included the preparation by the NRC staff of a Final Environmental Assessment following extensive public comment, the Met-Ed/GPU request was granted by a unanimous vote of the NRC Commissioners on June 12, 1980. The actual venting took place between June 28 and July 11, 1980.

This report contains a comprehensive discussion of the total effort included in the preparation and conduct of the TMI-2 Reactor Building venting program. Section 2.0 is devoted to examining the licensing efforts which were required to obtain the authorization to proceed with the venting. A description of the two systems utilized for venting, the Reactor Building and effluent radiation monitoring systems, and the computer program and procedures used to control the venting in conformance with Technical Specifications is presented in Section 3.0. Section 3.0 also contains a discussion of problems encountered during venting and a summary and analysis of pertinent venting data. The extensive radiological environmental monitoring programs including those of EPA (in conjunction with Pennsylvania State University), NRC, and Met-Ed/GPU and a Citizens Radiation Monitoring Program (in conjunction with the Pennsylvania Department of Natural Resources), are described and the important findings summarized in Section 4.0. References are extensively used in the text to identify the sources of information and to direct the interested reader to a more complete, detailed, or comprehensive discussion of the pertinent subject.

This report will be included in the Technical Integration Office (TIO) data bank which is being developed by the TMI-2 Information and Examination Program that was jointly established by the Department of Energy (DOE), the Nuclear Regulatory Commission (NRC), the Electric Power Research Institute (EPRI), and GPU to provide detailed, accurate technical documentation of research and development information and on-site cleanup activities at TMI-2.

2.0 LICENSING

By specific NRC order, NRC approval was required prior to processing the TMI-2 Reactor Building atmosphere. Before recommending for NRC approval decontamination of the Reactor Building by venting it to the environment, Met-Ed/GPU conducted a full evaluation of all available alternatives for removing Kr-85 including an examination of the need to remove the contained radioactive gas. Following the request by Met-Ed/GPU for NRC approval to release Kr-85 from the Reactor Building via controlled venting, an involved licensing effort commenced. The licensing process was complicated by the intense public interest and concern with the proposed Kr-85 venting. This section describes the licensing process that was required to obtain the necessary authorization to vent the Kr-85 contained in the Reactor Building. Section 2.1 presents the licensing chronology and Section 2.2 briefly summarizes the conclusions of the safety analyses and environmental assessments which were conducted. In Section 2.3, the technical specification changes granted by NRC as a part of their venting approval are presented.

2.1 Licensing Chronology

Following completion of the Met-Ed/GPU technical evaluation which concluded that venting was the best means of decontaminating the Reactor Building atmosphere, licensing became the critical path item to venting. The chronology of the Met-Ed/GPU - NRC licensing interactions is given in Table I. Approximately seven months transpired from the time of Met-Ed/GPU's submittal of the TMI-2 Reactor Building Purge Program Safety Analysis and Environmental Report and request for NRC's approval to vent (November 13, 1979) until the NRC Commissioners' approval was received (June 12, 1980) and venting started (June 23, 1980). The intervening activities consisted of Met-Ed/GPU responding to NRC requests for additional information, NRC preparation of an environmental assessment, public comments on the Draft Environmental Assessment and their resolution, special reviews by the State of Pennsylvania (UCS study) and the NRC Commissioners (Ertel study and SAI review of the Selective Absorption Process as an alternative to purging), and meetings and exchanges between the NRC staff, the Advisory Committee on Reactor Safeguards (ACRS), the NRC Commissioners, Met-Ed/GPU, and the public. Also during this period, all necessary procedures for venting were reviewed with the NRC.

TABLE I. MET-ED/GPU-NRC LICENSING CHRONOLOGY

November 13, 1979	Met-Ed/GPU submitted to NRC the Three Mile Island Unit 2 Reactor Building Purge Program Safety Analysis and Environmental Report and requested approval to proceed with purging the TMI-2 Reactor Building.[1]
December 18, 1979	NRC withheld approval to purge the TMI-2 Reactor Building pending preparation of an environmental assessment on this subject and the NRC Commissioners' approval of the specific method for disposition of the Kr-85 in the Reactor Building.[2]
	NRC, following their review of the Met-Ed/GPU Safety Analysis and Environment Report, requested additional information (33 questions) to complete their evaluation and prepare an environmental assessment.[3]
January 4, 1980	Met-Ed/GPU supplied responses to the NRC request for additional information (33 questions)[4] (Results of the Reactor Building air samples analyzed for Sr-89/90 were supplied later in References 5 and 6)
	Note: Following the above Met-Ed/GPU response to NRC's original 33 questions, NRC requested additional information (four questions) related to Reactor Building venting hardware concerns to which Met-Ed/GPU supplied responses.[7-11]
March 1980	NRC published the Draft Environmental Assessment for Decontamination of the Three Mile Island Unit 2 Reactor Building Atmosphere (NUREG-0662) and two subsequent Addenda for public comment.[12]

TABLE I (cont'd)

April 11, 1980	The NRC Commissioners and the Advisory Committee on Reactor Safeguards (ACRS) met. The ACRS members generally favored the expeditious decontamination of the Reactor Building atmosphere by controlled purging to the environment.
May 1980	NRC published the Final Environmental Assessment for Decontamination of the Three Mile Island Unit 2 Reactor Building (NUREG-0662) in which the NRC staff recommended a "slow purge" of Kr-85 from the TMI-2 Reactor Building.[13]
June 4, 1980	Met-Ed/GPU in a letter to NRC stated and justified its intent to remove the cap on the plant ventilation stack and to discontinue use of the supplemental ventilation system atop the Auxiliary Building.[14]
June 5 & 10, 1980	NRC Commissioners briefings by the NRC staff on the venting of Kr-85 from the TMI-2 Reactor Building.
June 12, 1980	The NRC, in Memorandum and Order CLI-80-25, gave approval for Met-Ed/GPU to purge the TMI-2 Reactor Building atmosphere. An Order for Temporary Modification of License for the period of the purge and a Negative Declaration concerning the need for an environmental impact statement (EIS) were also issued. (Note: In the matter of the requirement for an EIS, on May 19, 1980 the President's Council on Environmental Quality said the NRC staff proposal to separate the decontamination of the Reactor Building atmosphere from the preparation of a programmatic EIS on decontamination and disposal of radioactive wastes resulting from the TMI-2 March 28, 1979 accident did not violate regulations implementing the National Environmental Policy Act.)

TABLE I (cont'd)

June 23, 1980	Met-Ed/GPU requested a Technical Specification change to bypass a Reactor Building purge exhaust interlock because it had been superseded by the above NRC Order for Temporary Modification of License.[15]
	A joint motion for reconsideration of the NRC's June 12, 1980 Memorandum and Order and Order for Temporary Modification of License was filed by Steven C. Sholly, the Newberry Township Three Mile Island Steering Committee, and People Against Nuclear Energy ("PANE"). They cited the findings of the "Heidelberg" study[16] as one of their four principal arguments for reconsideration.
June 24, 1980	NRC issued an amendment to the Met-Ed/GPU license for TMI-2 as requested in Met-Ed/GPU's June 23, 1980 letter above.[17]
June 26, 1980	The NRC denied the joint motion filed June 23, 1980 for reconsideration of the NRC's June 12, 1980 Memorandum and Order and Order for Temporary Modification of License.
June 27, 1980	NRC approved the "Unit #2 Operating Procedure 2104-4.82 Reactor Building Atmosphere Cleanup Using the Modified Hydrogen Control System and the "B" Train of the Modified Reactor Building Purge System" and the modifications to the hydrogen control and Reactor Building purge systems for purging the TMI-2 Reactor Building.[18]
June 28, 1980	TMI-2 Reactor Building venting commenced.

TABLE I (cont'd)

July 3, 1980	Steven C. Sholly moved the Atomic Safety and Licensing Board for TMI-2 to order suspension of venting pending the completion of hearings on the matter of venting Kr-85 from the TMI-2 Reactor Building. (Note: Sholly subsequently dropped this motion.)
July 11, 1980	TMI-2 Reactor Building venting was completed.

The need for an Environmental Assessment by NRC of the proposed Reactor Building venting plan was due to the desire to vent the Reactor Building prior to the completion of the programmatic environmental impact statement on decontamination and disposition of radioactive waste resulting from the March 28, 1979 accident at TMI-2. This programmatic environmental impact statement was required by the NRC's Statement of Policy dated November 21, 1979. While this Statement of Policy provided that decontamination action prior to the completion of the programmatic statement was not precluded, it stated in particular that purging the Reactor Building of radioactive gases could not occur without a prior environmental review and the opportunity for public comment.

The involvement of various federal, state, and local agencies and officials, of non-governmental organizations, and of private individuals in the licensing process was principally through their comments on the NRC Draft Environmental Assessment.[12] NRC published the Draft Environmental Assessment in March 1980 and two subsequent addenda for public comment. The extended public comment period ended May 16, 1980 and a Final Environmental Assessment[13] was completed that same month. At the close of the comment period approximately 800 responses had been received. A number of separate reports were also generated by independent organizations and submitted as part of the comment process. All substantive comments received are contained in Volume 2 of the Final Environmental Assessment, and Section 9.0 of Volume 1 of the Final Environmental Assessment provides NRC's responses to these comments.

Of special interest was the establishment of a "Blue Ribbon Panel" by Pennsylvania Governor Thornburgh to independently evaluate venting the TMI-2 Reactor Building. The panel was composed of members of the Union of Concerned Scientists. Both NRC and Met-Ed/GPU personnel met with and supplied requested information to the Union of Concerned Scientists on plant status and the need to purge the Reactor Building. The Union of Concerned Scientists' report,[19] although it recommended consideration of two alternative venting plans, did conclude that there would be no direct radiation-induced health effects associated with the proposed venting plan (see Section 2.2.2). Based on this and several other similar requested reports, Governor Thornburgh, in a letter to NRC Chairman John Ahearne, said he was "prepared to support venting of Kr-85" from TMI-2, based on a "broad consensus" that the process "is, indeed, a safe one."

The NRC Commissioners also became actively involved in the Environmental Assessment review particularly with the question of the viability of the Selective Absorption Process as an alternative to venting. To obtain information first hand, Pennsylvania Congressman Ertel and Commissioner Gilinsky visited Oak Ridge National Laboratory (ORNL) in April 1980 where the Selective Absorption Process has been developed. Additionally, at the request of the Commission, the NRC Office of Policy Evaluation (a Commission staff office) contracted with SAI to perform an independent technical evaluation of the Selective Absorption Process as a purging alternative. Although Congressman Ertel believed the Selective Absorption Process could be placed into operation in six months, ORNL itself believed 13 months was a "best efforts" estimate with others concluding 16 months or even longer was optimistic. The SAI study[20] found purging to be the best alternative when compared to the Selective Absorption Process from all points of consideration including feasibility, effectiveness, practicality, health and safety, psychological stress on nearby population, schedule and cost.

One activity which consumed considerable energies but which is not shown in Table I was the public information efforts by Met-Ed/GPU, NRC, EPA, and the Pennsylvania Department of Environmental Resources (DER). These efforts to better inform the public in the area around TMI about the proposed Reactor Building venting plan and its radiological environmental impact were extensive. For example, to educate the public with regards to the contents of the Draft Environmental Assessment, NRC, generally accompanied by members of EPA and DER, participated in 15 public meetings and meetings with interested citizens groups, 16 meetings with elected officials, and seven press conferences and appearances on public information radio and television shows. NRC also published an easy-to-understand report entitled "Answers to Questions about Removing Krypton from the TMI-2 Reactor Building" (NUREG-0673). Met-Ed/GPU participated with NRC in the above meetings and public appearances along with others including a live hour-long community service broadcast just prior to venting. Met-Ed/GPU also provided a special telephone information center before and during venting where people could call and ask questions and receive answers about the venting program. A similar telephone service was available to the news media which Met-Ed/GPU also provided with a special venting briefing package.

The licensing efforts shown in Table I also do not include subsequent civil court actions which followed the NRC's June 12, 1980 Memorandum and Order and Order for

Temporary Modification of License. On June 23, 1980 a petition was filed in the United States Court of Appeals for the District of Columbia Circuit seeking to review the NRC Orders of June 12, 1980. (Steven Sholly and Donald E. Hossler versus the U. S. NRC, Chairman Ahearne and the other Commissioners as individuals.) The petitioners sought to enjoin the scheduled venting pending 30 day's public notice and the opportunity for a hearing prior to the commencement of venting. On June 26, 1980 the Court denied the petitioners' motion for injunctive relief pending appeal. On June 27, 1980 the Court denied a petition for rehearing and a suggestion for rehearing en banc (by the entire court). On June 28, 1980 the Court denied the petitioners' further motions for a five day injunction pendente lite (during the course of litigation) and for a writ of mandamus. Even following the commencement of venting on June 28, 1980, People Against Nuclear Power (PANE) represented by Steven Sholly and Donald E. Hossler (PANE's president) filed a petition in the United States Court of Appeals for the Third Circuit raising the same issues and seeking the same relief as in the previous District of Columbia Circuit petition. On July 10, 1980 the Third Circuit transferred the PANE petition and other papers to the District of Columbia Court. On July 11, 1980 when the venting of Kr-85 from the Reactor Building was completed the Sholly et al V. NRC et al was still before the Court and although still pursued by the petitioners for other reasons, was moot insofar as the venting activities were concerned. (Note: In its later decision, the Court finally ruled in favor of the plaintiff that a public hearing should have been held prior to venting.)

As a final note to this section, it should be pointed out that at their first Environmental Assessment review the NRC Commissioners requested the NRC staff consider the use of more rapid venting. The advantage to completing the venting as rapidly as possible was the minimization of the public psychological impact (see Section 2.2.2). Rapid venting was possible with the Reactor Building Air Purge and Purification System, but was not part of the original Met-Ed/GPU venting proposal because it required the temporary waiving of the existing Technical Specifications on radioactive material releases. Based on the NRC staff review, the final NRC order permited rapid venting through a temporary Technical Specification change allowing the venting to be accomplished as quickly as possible.

2.2 Safety Analysis and Environmental Assessment

The complete scfety analysis and environmental assessment of the decontamination of the TMI-2 Reactor Building atmosphere is contained in the November 12, 1979 Met-Ed/GPU Safety Analysis and Environment Report, the ensuing responses to NRC staff requests for additional information, and the NRC's Final Environmental Assessment. A brief summary of the major findings of the Met-Ed/GPU and NRC safety analyses and environmental assessments are presented below.

2.2.1 Need for Decontamination of the Reactor Building Atmosphere

Met-Ed/GPU in the Reactor Building Purge Program Safety Analysis and Environmental Report justified the need to decontaminate the Reactor Building atmosphere because without decontamination, the Reactor Building entry program and the effectiveness of operations toward ultimate fuel removal would be significantly complicated and restricted. Met-Ed/GPU also stated that leaving the Kr-85 in the Reactor Building atmosphere while other steps toward fuel removal proceeded represented a substantial risk of ultimate uncontrolled release of Kr-85 to the environment and an unacceptable increase in operation and cleanup personnel exposure.

The NRC staff in the Final Environment Assessment agreed that the Reactor Building atmosphere needed to be decontaminated in a timely manner primarily to permit the less restricted access to the Reactor Building necessary to gather information, to maintain instruments and equipment, and to proceed toward total decontamination of the TMI-2 facility. Specifically, NRC concluded that delaying the removal of the Kr-85 from the Reactor Building atmosphere would have meant:

- Added difficulty and risks to workers who entered the Reactor Building since they would have been required to wear heavy protective clothing and air-supply equipment.

- Increased operation personnel radiation exposure.

- Interference with needed maintenance of equipment in the Reactor Building whose failure could effect the ability to maintain the safe condition of the reactor core or whose failure might cause leakage of radioactive material from the Reactor Building.

- Increased risks of uncontrolled release of radioactive material to the environment.

- Increased anxiety and stress to the surrounding public because of the indecisive management of the Reactor Building atmosphere decontamination and the increased possibility of uncontrolled releases of radioactive material.

Both Met-Ed/GPU and NRC agreed also that until the ultimate removal of the fuel in the reactor, there exists a small but finite potential for inadvertent core recriticality. Hence, there was an immediate and justifiable need to remove the Kr-85 existing in the Reactor Building in order to proceed with the safe and expeditious completion of all cleanup activities at TMI-2 and to reduce the potential for unpredictable and uncontrollable radioactive material leaks to the environment.

2.2.2 Health Effects, Psychological Stress, and Accidents

Met-Ed/GPU and NRC both concluded that there would be negligible physical health effects associated with the properly controlled venting of the estimated 57,000 curies of Kr-85 from the Reactor Building. This conclusion was supported by others including the U. S. Environmental Protection Agency (EPA), the U. S. Department of Health and Human Services (formerly HEW), the National Council on Radiation Protection and Measurements, the Pennsylvania Department of Environmental Resources (DER), the U. S. Department of Energy, the National Resources Defense Council, and the Union of Concerned Scientists. NRC predicted the total off-site dose to the maximum exposed individual would be 11 mrem beta skin dose and 0.2 mrem total body gamma dose. Met-Ed/GPU calculations estimated roughly one-half of these exposures. For the collective surrounding 50-mile off-site population of 2.2 million people, doses were predicted by NRC to be 0.76 and 63 person-rem for total body and skin doses, respectively. Similar doses were calculated by Met-Ed/GPU. NRC estimated that these person-rem doses could cause:

Total potential cancer deaths	0.0001
Genetic abnormalities	0.0002
Skin cancer deaths	0.000006

These numbers are insignificant fractions of the number of cancer deaths and genetic abnormalities that will occur in the surrounding 50 mile population of 2.2 million from all other factors.

Kr-85 venting was also to occur only during acceptable meteorological conditions and be controlled to remain within the limits established by 10 CFR Part 20, the design objectives of 10 CFR Part 50, Appendix I, and the provisions of 40 CFR Part 190.0, to the extent they may be applicable. Conformance with these limits would be confirmed by extensive off-site radiological monitoring conducted by EPA, a Citizen Radiation Monitoring Program, NRC, Pennsylvania State University, and Met-Ed/GPU (see Section 4.0).

An occupational exposure of 1.2 person-rem was estimated by NRC to result from the Reactor Building venting operation. This exposure was by far the least for all the Kr-85 removal alternatives considered (see Section 2.2.3).

With respect to psychological stress, NRC concluded after consultation with expert psychologists, that the resulting stress from the plan to vent Kr-85 would be less than any of the alternative plans considered (see Section 2.2.3). Venting the Reactor Building was also believed to have the net effect of reducing the stress which otherwise would occur if positive steps were not taken promptly to proceed with decontamination and reduce uncertainty about the present and future condition of TMI-2. It was recognized, however, that venting Kr-85 might be unpopular to certain segments of the local population.

A conservative analysis by NRC of the worst case accident which could occur during venting resulted in a whole body gamma dose to an individual at the site boundary of 0.3 mrem and a beta skin dose of 25 mrem. This total body dose represents only a small fraction of the 10 CFR Part 100 limit of 25 rem. (Skin dose limits are not included in 10 CFR Part 100.)

2.2.3 Alternatives

In addition to the proposed venting plan, a number of alternatives which could reduce off-site radiation exposure even further were considered and evaluated by Met-Ed/GPU prior to selection of venting as the best and safest method. These alternatives included:

1. No action
2. Selective absorption
3. Charcoal absorption, including a refrigerated absorber system
4. Gas compression and storage
5. Cryogenic processing (liquifying the gas and storing for later disposal)

NRC's consideration of alternatives included all of the above plus:

1. Venting, but at a faster rate or elevating or heating the release to obtain better atmospheric dispersion
2. A combination of venting and the other alternatives

The no action alternative was dismissed for the reasons described in Section 2.2.1 concerning the need for prompt removal of the Kr-85 from the Reactor Building. Each of the other proposed alternatives underwent a thorough examination by first defining a workable alternative system and then evaluating it in terms of resulting off-site radiation exposures, occupational radiation exposures, potential accident consequences, timeliness, and cost. For all alternatives it was found that the expected occupational radiation exposure and postulated maximum credible accident consequences were equal to or higher than for the proposed "slow purge" method. It was also determined that none of the alternatives could be implemented in the near future. Thus, the further reduction by these alternatives of the already negligible environmental impact from the slow purge option was not deemed significant enough to outweigh their increased occupational exposures, more severe accident consequences, and untimeliness. Hence, the NRC staff concurred with Met-Ed/GPU's choice of purging as the quickest and safest plan.

2.3 Technical Specifications

Accompanying the NRC's June 12, 1980 Memorandum and Order authorizing the venting, NRC issued an Order for Temporary Modification of License which revised Section 2.1.2 of Appendix B of the TMI-2 Technical Specifications The change temporarily suspended the instantaneous and quarterly limits for releases of noble gases (i.e., Kr-85) and replaced them with equivalent off-site dose limits. Table 2 presents the text of this change to the Technical Specifications.

One additional temporary change to the TMI-2 Technical Specifications was requested and received as a result of the above suspension of release limits for Kr-85. This change bypassed the radiation interlock on the Reactor Building Purge and Purification System exhaust which automatically repositioned the system dampers from the open to the recirculation mode if the monitored instantaneous release rate of radioactive material exceeded a given value. The text of this revision to the Technical Specifications is presented in Table 3.

The NRC's June 12, 1980 venting authorization order also directed that Met-Ed/GPU conduct the venting in accordance with procedures approved by the NRC, pursuant to Section 6.8.2 of proposed Appendix A to the Technical Specifications as made binding by the February 11, 1980 order of the Director, Office of Nuclear Reactor Regulation. Compliance with this requirement was met when the NRC in its June 27, 1980 letter approved the "Unit #2 Operating Procedure 2104-4.82 Reactor Building Atmosphere Cleanup Using the Modified Hydrogen Control System and the "B" Train of the Modified Reactor Building Purge System." Subsequent revisions to this procedure during the venting process were also approved by the NRC.

TABLE 2. <u>TEMPORARY TECHNICAL SPECIFICATION CHANGE REQUIRED FOR Kr-85 VENTING</u>

Only for the period of the purge of the TMI-2 Reactor Building atmosphere, Section 2.1.2h is deleted and Sections 2.1.2a and 2.1.2c are superseded by the following:

Do not exceed for the maximally exposed individual* in any one of the 16 (22 1/2°) sectors centered on the TMI-2 Reactor Building any of the following:

(a) 15 mrem skin dose

(b) 5 mrem total body dose

(c) 20% of the limits in (a) and (b) shall not be exceeded over any one hour period

In addition, pursuant to Section 6.8.2 of the proposed Appendix A Technical Specifications, NUREG-0432, made binding on the licensees by the February 11, 1980 order of the Director of the Office of Nuclear Reactor Regulation (NRR), any purging shall be conducted in accordance with procedures approved by the Director, NRR.

Under the above conditions, the licensee is to minimize the total time required to complete purging the Reactor Building to 10 CFR Part 20 MPC (for workers).

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*Maximally Exposed Individual

(1) One hypothetical individual within <u>each</u> of 16 sectors at off-site location with maximum anticipated dose.

(2) No allowance for occupancy time - assume individual present continuously.

(3) No hypothetical individual shall receive more than dose design objectives of (a) and (b) above.

TABLE 3. TEMPORARY TECHNICAL SPECIFICATION CHANGE REQUIRED FOR Kr-85 VENTING

Only for the period of the purge of the TMI-2 Reactor Building atmosphere pursuant to the Commission's Order for Temporary Modification of License dated June 12, 1980, Section 2.1.2B.3 of the Appendix B Technical Specifications is superseded by the following:

> Unit 1 valves AH-V1A and AH-V1B shall be interlocked to close or recirculate on receipt of a high radiation signal from the Reactor Building Exhaust Monitor RM-A9. The interlocks from the Unit 2 Reactor Building Exhaust Monitors HP-R-225 and HP-R-226 which initiate closure or recirculation of the Unit 2 Dampers D5129 A/D and D5129 B/C may be bypassed in accordance with procedures approved pursuant to Appendix A Technical Specification 6.8.2.

3.0 REACTOR BUILDING VENTING

This section covers all aspects of the decontamination of the TMI-2 Reactor Building atmosphere accomplished by venting the contained Kr-85 to the environment except for the associated environmental radiation monitoring programs which are discussed in Section 4.0. After a brief introduction where the Reactor Building source term is defined, this section includes a description of the venting systems, equipment, and instrumentation; a description of how the venting was controlled within established Technical Specification requirements; how the venting operation was conducted; the venting chronology and problems which arose; a summary of the venting data and results; and finally, an analysis of the venting data and results.

3.1 Reactor Building Source Term

The Reactor Building atmosphere source term determination principally involved three types of Reactor Building air samples: noble gas, particulate matter, and radioiodine. In addition, samples for tritium and gross beta determination were also obtained. The equipment used to collect Reactor Building air samples is discussed in Section 3.2.3 below. Reference 21 provides an even more extensive discussion of the determination by Met-Ed/GPU of the Reactor Building atmospheric radioactive material content and chemical composition and the evolution of equipment and sampling procedures following the March 28, 1979 accident.

Prior to June 28, 1980 when venting began, sample results showed the Kr-85 level at 1.04 μCi/cc. All other noble gases (i.e., Xe-131m, Xe-133m, Xe-133, and Xe-135) had decayed to below minimum detectable activity (MDA) levels of 1 E-6 μCi/cc. Based on an estimated free volume of the Reactor Building of 2 E6 ft^3 (2,131,178 ft^3 from Reference 22, minus the volume of water in the basement), there was a Reactor Building inventory of about 57,000 curies of Kr-85. (Note: This Kr-85 inventory was later revised downward, see Section 3.7.)

Radioactive decay had also reduced iodine levels in the Reactor Building at the time of venting to below MDA levels of 1 E-9 μCi/cc. Particulate levels, primarily Cs-137, were less than 1 E-9 μCi/cc. Specific analyses of five samples for Sr-89/90 showed Sr-89 airborne particulate activity ranging from 5 E-11 μCi/cc to 8 E-10

μ Ci/cc and Sr-90 from 9 E-11 μCi/cc to 7 E-10 μCi/cc. Gross beta-gamma airborne particulate activities ranging from 5 E-8 μCi/cc to 9 E-10 μCi/cc also indicated very little Sr-89/90 was airborne. The airborne concentration levels of all the above isotopes are below the maximum permissible concentration levels listed in Table I of Appendix B to 10 CFR Part 20. Because of these low levels and because all Reactor Building venting would be through high efficiency HEPA filters (see Sections 3.2.1 and 3.2.2), essentially no release of particulate radiation was expected.

Airborne tritium concentrations in the Reactor Building prior to venting were measured at between 10^{-4} and 10^{-5} μCi/cc.

In addition to the above results obtained by Met-Ed/GPU, Table 4 presents the results of samples taken for the Technical Integration Office as a part of the TMI-2 Information and Examination Program during the period of April 29 to May 2, 1980 using the glove box and sampling apparatus installed in containment penetration R-626. These samples taken at R-626 underwent a more sophisticated analysis than utilized by Met-Ed/GPU, thus providing additional information on radioisotope concentrations in the Reactor Building atmosphere prior to venting.[23]

As can be seen from the above Reactor Building source term data, Kr-85 was by far the dominant radioactive isotope and was the limiting isotope relative to controlling venting flow rates. These data, especially the Table 4 data, were also used by the NRC to refute the Heidelberg report[16] finding (see Section 2.1) that radionuclides such as C-14, Co-60, Sr-89, Sr-90, Ru-106, Cs-134, Cs-137, Pu-239, and Pu-241 could lead to radiation exposures significantly higher than those caused by Kr-85 if purging the Reactor Building occurred.

3.2 Venting Systems, Equipment, and Instrumentation

The decontamination of the TMI-2 Reactor Building atmosphere was accomplished using two existing systems to purge or vent the contained Kr-85 to the environment although some modifications were required. The two existing systems, the Hydrogen Control System and the Reactor Building Air Purge and Purification System, are really subsystems of the Reactor Building Ventilation and Purge System which also includes the Heat Removal (Reactor Building Air Cooling) subsystem. The Modified Hydrogen Control System was capable of venting at rates from 0 to 600 cfm.

TABLE 4. RADIONUCLIDE CONCENTRATIONS IN THE TMI UNIT-2 REACTOR BUILDING ATMOSPHERE

April 29 - May 2, 1980

Isotope	$\mu Ci/cm^3$ [a]	Isotope	$\mu Ci/cm^3$ [a]
^{3}H	4.7 ± 0.8 E-5	^{106}Rh	<1 E-10
^{14}C	3.5 ± 0.9 E-7	^{110m}Ag	1.6 ± 0.6 E-11
^{51}Cr	<6 E-10	^{124}Sb	<1 E-10
^{54}Mn	2 ± 2 E-11	^{125}Sb	<2 E-10
^{55}Fe	<5 E-11	^{129m}Te	4 ± 2 E-10
^{59}Fe	<3 E-11	^{129}I	6.6 ± 0.5 E-11
^{57}Co	<1 E-11	^{134}Cs	1.3 ± 0.1 E-10
^{58}Co	1.0 ± 0.3 E-11	^{137}Cs	8.4 ± 0.9 E-10
^{60}Co	3 ± 2 E-12	^{141}Ce	<6 E-11
^{63}Ni	<2 E-11	^{144}Ce	<9 E-11
^{85}Kr	1.02 ± 0.05 [b]	^{152}Eu	<3 E-11
^{89}Sr	7 ± 3 E-11	^{154}Eu	<2 E-11
^{90}Sr	1.9 ± 0.3 E-10	^{155}Eu	<3 E-11
^{91}Y	<3 E-11	^{235}U	<7 E-13
^{95}Zr	<2 E-11	^{238}U	<7 E-13
^{103}Ru	<4 E-11	^{238}Pu	<7 E-12
		$^{239/240}Pu$	<2 E-12

a. Volume units are cm^3 at STP.

b. Conversion to actual RB conditions from STP yields 0.88 ± 0.04 $\mu Ci/cm^3$.

Venting at rates above 600 cfm and on up to 18,500 cfm was done with the Modified "B" train of the Reactor Building Air Purge and Purification System. The allowed venting rate and hence the selection of which system was used was a function of the current meteorology and Reactor Building atmosphere Kr-85 concentration (see Section 3.3).

The Modified Hydrogen Control System and the Modified "B" train of the Reactor Building Air Purge and Purification System are described in Sections 3.2.1 and 3.2.2, respectively. Reactor Building and effluent radiation monitoring equipment utilized during venting are discussed in Section 3.2.3. Additional equipment and modifications supportive of the venting program are addressed in Section 3.2.4.

3.2.1 Modified Hydrogen Control System

The TMI-2 Hydrogen Control System (HCS) was originally designed for use as a back-up for the hydrogen recombiner to maintain the Reactor Building hydrogen concentration below combustion limits following a loss of coolant accident (LOCA). This was achieved by drawing Reactor Building air through a filter train with an exhaust fan and then discharging it to the environment through the plant vent stack (160 feet above grade level). The HCS was modified for the Reactor Building venting (see Table 5) to become the Modified Hydrogen Control System (MHCS), a complete description of which is provided below.

3.2.1.1 System Description. The MHCS located in the Auxiliary Building is shown in Figure 1. The MHCS draws Reactor Building air via containment isolation valves AH-V3A and AH-V52 through a filter train with an exhaust fan which discharges to the station vent. The filter train consists of a prefilter, a HEPA filter, an activated carbon filter, and another HEPA filter. The original HCS exhaust fan (150 cfm capacity) was replaced with a larger fan to increase the design flow rate of the MHCS up to the filter train capacity (1000 cfm). In the original HCS design, the hydrogen purge flow rate was controlled by throttle valve AH-V25. In the MHCS, valve AH-V36 was modified to provide fine control of the flow rate over the full, increased range of flow. Replacement air to the Reactor Building was supplied through valves AH-V7 and AH-V3B and controlled by opening and closing AH-V7 to maintain the Reactor Building pressure between -0.5 and -0.1 inches of Hg. Instrumentation was provided to indicate, record, and/or alarm filter train differential

TABLE 5. HYDROGEN CONTROL SYSTEM MODIFICATIONS

In order to use the Hydrogen Control System to safely vent the Reactor Building at rates up to 600 cfm, the following modifications were made:

- Replaced the existing fan, AH-E-34, with a fan capable of at least 600 cfm flow rate.

- Provided new instrumentation to measure the increased MHCS flow rate.

- Added manual jog control to valve AH-V36 and a 30 second time delay to close AH-V36 upon exhaust fan shutdown. Remote control of AH-V36 was also provided from the Control Room on a new panel located adjacent to Panel No. 25.

- Provided an interlock to close AH-V7 on loss of power to the MHCS exhaust fan. Remote control of AH-V7 was also provided from the Control Room on a new panel located adjacent to Panel No. 25.

- Provided interlocks to trip the MHCS exhaust fan on high activity as measured on HPR-229, on failure or loss of power to HPR-229, or on loss of instrument air to AH-V36.

TABLE 5 (cont'd)

- Provided interlocks to close AH-V3A and B on high Reactor Building pressure.

- Provided a gamma monitor probe in the hydrogen control filter housing to monitor the buildup of radioactive material on the filters.

- Replaced the HPR-229 isokinetic probe tip in order to get accurate readings with the new increased flow rates and added a high range gas channel to measure radiation levels up to 1000 μCi/cc.

- Added five electric infrared type radiant heaters along the outside of the MHCS filter plenum to ensure that moisture formation, which could decrease the particulate removal efficiency of the HEPA filters, did not occur.

- Provided an interlock to shutdown the MHCS exhaust fan on high MHCS filter housing vacuum.

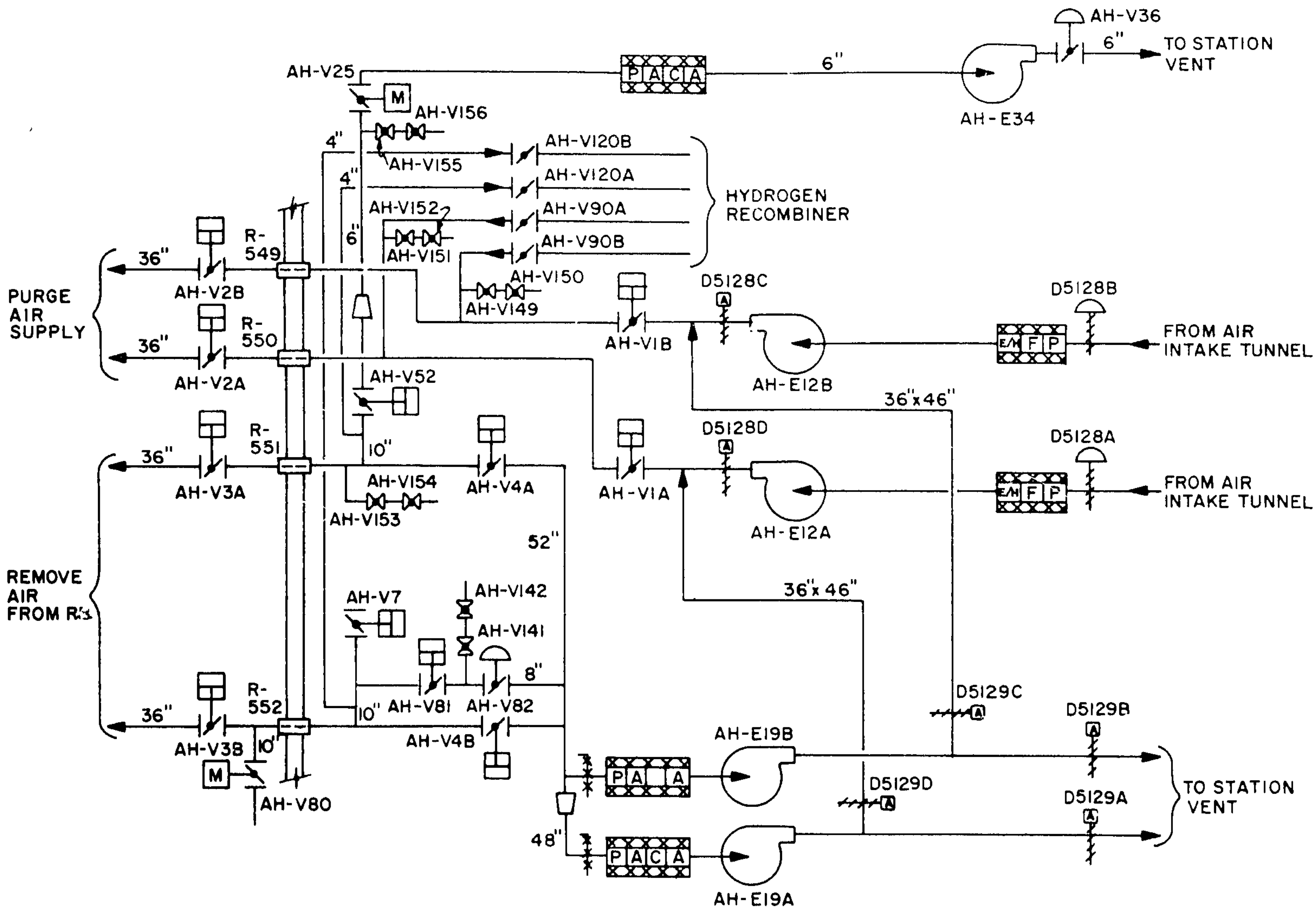

Figure 1 TMI-2 Purge System.

pressure, exhaust flow rate, and effluent radiation levels. Interlocks were included to protect MHCS equipment and for rapid isolation on equipment failure or high radiation levels at the fan discharge. The HCS was designed for 30 psig and 150°F and seismic Class I conditions. The system also meets the requirements of the ANSI B31.0 Code for Power Piping.

Because make-up air to the Reactor Building was supplied through AH-V7 and AH-V3B, a potential recirculation problem existed between this Reactor Building make-up air line and the MHCS exhaust line. These lines are both exhaust lines for the Reactor Building Air Purge and Purification System (RBPPS) and are only about three feet apart inside the Reactor Building. This situation was unavoidable, however, because the only other alternative was to open up a 36" RBPPS supply line for make-up air (the make-up air line to AH-V7 is only a 10" line). This had been ruled out to minimize the decrease in containment integrity during the purge. A similar problem did not exist for the MRBPS since it utilized the normal RBPPS "B" train supply and exhaust lines which open up outside and inside the Reactor Building D-ring, respectively, preventing any possibility for direct recirculation. To help insure good Reactor Building atmosphere mixing when operating either venting system, but especially the MHCS, the Reactor Building air cooling fans were operated continuously during venting. Even their effectiveness may have been hampered, however, because at the lower Reactor Building levels at least some of the ventilation ducts were partially under water.

3.2.1.2 Component Description.

MHCS Exhaust Unit--The MHCS exhaust unit is located in the Auxiliary Building at an elevation of 328 feet. The unit is comprised of a bank of filters housed in a steel cabinet and an exhaust fan connected to the housing. The filter bank consists of the following filters listed as they occur in the flow path:

(a) Pre-Filter AH-F-36
(b) HEPA Filter AH-F-33
(c) Activated Carbon Filter AH-F-34
(d) HEPA Filter AH-F-35

Access doors are located on top of the housing for easy maintenance. There is a differential pressure switch connected across the filter bank which will initiate an alarm in the Control Room (Panel No. 25) on high differential pressure (setpoint 6.0" W.G. at 1000 cfm).

A specific concern raised by NRC during their licensing review of the Met-Ed/GPU proposed venting system was the potential for filter housing failure and resultant radioactive material leakage because the filter housing was designed to withstand only 18 inches W.G. (water gauge) vacuum and the MHCS exhaust fan was capable of producing higher negative pressures in the filter plenum. Resolution of this disparity was achieved by installation of redundant pressure switches to shutdown the exhaust fan when the pressure in the filter housing reached 15 inches W.G. vacuum.

MHCS Pre-Filter AH-F-36--The pre-filter is a replaceable bag filter designed for rough particle removal (see Table 6). It has a local differential pressure indicator.

MHCS Absolute (HEPA) Filters AH-F-33 and AH-F-35--The HEPA filters (Table 6) are constructed of a dry fibrous high interception, sub-micron glass fiber which has an efficiency of 99.97% for particles larger than 0.3 microns. The filters conform to ORNL-NSIC-65. The filters are mounted in a steel frame and have aluminum separators. Each HEPA filter is fitted with a local differential pressure indicator.

MHCS Activated Carbon Filter AH-F-34--The activated carbon filter is designed to trap and remove gaseous contaminants (iodine) from the airstream. The carbon filter (Table 6) is of activated charcoal impregnated type, and is of water repellant and fire resistant construction. The adsorbent material (MSA 85851) is housed in a stainless steel flat bed type frame. The filter is tested in accordance with ORNL-NSIC-65.

The carbon filter is instrumented with a local differential pressure indicator. A fire detector and an automatic deluge system are provided for fire protection of the carbon filter bank. Means for detecting radiation levels and leaks are provided through a flanged rubber sock-port opening at the upstream and downstream face of each filter bank where radiation monitor probes can be inserted.

TABLE 6. MHCS EXHAUST FAN AND FILTER TRAIN
DESIGN PERFORMANCE AND EQUIPMENT DATA

MHCS Exhaust Fan	
Quantity	1
Type	Centrifugal Exhauster with Direct Drive
Flow, cfm	0 to 1000
Static Pressure, in W.G.	48 neg at 3550 rpm
Fan (Motor) Speed, rpm	3550
Fan Motor Voltage/No. of Phases/Hz	460/3/60
Motor H.P.	15
MHCS Filter Train	
Prefilter - AH-F-36	
Quantity	1
Type	Disposable bag filter
Clean Pressure drop, in W.G.	0.8
Max. Capacity, cfm	1000
Face Velocity through Filter, fpm (max.)	500
Size of Filter, inches	24x24x36
Efficiency	93% (NBS Dust Spot Method)
Seismic Classification	I

TABLE 6 (cont'd)

Absolute (HEPA) Filters - AH-F-33 and -35

Quantity	1
Clean Pressure drop, in W.G.	1.0
Max. Capacity, cfm	1000
Size of Filter, inches	24x24x11-1/2
Efficiency	99.97% for particles larger than 0.3 microns
Seismic Classification	I

Carbon Filter - AH-F-34

No. of Cells	3
Type	Flat-bed radioactive iodine adsorption activated carbon
Max. Capacity, cfm	1000
Flow through cell, cfm	333
Clean Pressure drop, in W.G.	1.0
Size of Filters, inches	24x40x7-3/4
Efficiency	99.9% of radioactive iodine in vapor form (Freon-112, 0.05 ppm by volume)
Seismic Classification	I

MHCS Exhaust Fan, AH-E-34--The original Hydrogen Control System exhaust fan was replaced by a 1000 cfm capacity fan manufactured by Buffalo Forge Company. The MHCS exhaust fan, located on the 328 foot level of the Auxiliary Building, was driven by a Westinghouse, 15 horsepower motor (see Table 6). However, when the new HCS exhaust fan was installed and tested, it was found that its exhaust flow rate was limited to 600 cfm by duct sizing.

The MHCS exhaust fan motor could be powered from two different power sources. Each power source had an "on-off" switch located on Panel No. 25 in the Control Room. There were red lights to indicate which of the two sources were lined up to power the fan motor and its associated valves (AH-V-25, 36, and 52). Two PULL-TO-LOCK-STOP-NORMAL-START switches were located on Panel No. 25 for the exhaust fan motor, one for each of the two power supplies. Additionally, the fan motor had a local START/STOP pushbutton. MHCS exhaust fan run indication was available on Panel No. 25 and locally.

Valve AH-V7--An air cylinder operated, ten inch carbon steel butterfly valve with an ANSI rating of 100 psig and a design temperature of 300°F is located in a branch connection off the Reactor Building purge exhaust line between Reactor Building penetration R-552 and the outer isolation valve AH-V4B, on the 328 foot level of the Auxiliary Building. The valve is in full compliance with the "Draft ASME Code for Pumps and Valves for Nuclear Power," Section B, Nuclear Class II Valves. The valve fails closed with a loss of instrument air. The valve is normally locked closed with its outlet flow path blanked off. As part of the MHCS, AH-V7 was unlocked and the outlet flow path opened. in addition, the original local control of the valve was changed so that the valve could be operated from the Control Room on a panel built adjacent to Panel No. 25.

Valve AH-V25--A motor operated six inch, carbon steel, butterfly valve with ANSI rating of 150 psig and a design temperature of 150°F is located in the hydrogen control line upstream of the MHCS exhaust fan. The valve and exhaust fan receive their power from the same source. The source is determined by power selection switches on Panel No. 25. The valve must be partially open (greater than 20%) for the fan to operate. The valve is controlled and has position indication both locally and at Panel No. 25.

Valve AH-V36--A diaphragm operated, six inch carbon steel butterfly valve with an ANSI rating of 150 psig and a design temperature of 150°F is located in the hydrogen control discharge line. The valve fails closed with a loss of instrument air. AH-V36 is on the 328 foot level of the Auxiliary Building. One of the modifications to the HCS was to provide fine motion control for AH-V36 to control the MHCS flow rate. Operation of this valve was by a jog switch with a 0-100% readout located on the panel constructed in the Control Room next to Panel No. 25.

Valve AH-V52--An air cylinder operated, ten inch carbon steel valve with an ANSI rating of 100 psig and a design temperature of 300°F is located in the hydrogen control line upstream of valve AH-V25. The valve is in full compliance with the "Draft ASME Code for Pumps and Valves for Nuclear Power," Section B, Nuclear Class II Valves. This containment isolation valve is normally padlocked shut and is only opened for hydrogen exhaust fan operation. The power source is similar to that described for AH-V25. The valve fails closed with loss of instrument air. AH-V52 is on the 328 foot level of the Auxiliary Building.

Valves AH-V3A and B--An air cylinder operated, 36 inch carbon steel butterfly valve with an ANSI rating of 100 psig and a design temperature of 300°F is provided in both the Reactor Building Air Purge and Purification System exhaust lines inside the Reactor Building on the 305 level. AH-V3A was the inner containment isolation valve on the line used by the MHCS to draw air from the Reactor Building and AH-V3B was the inner containment isolation valve used to supply replacement air to the Reactor Building. The valves are in full compliance with "Draft ASME Code for Pumps and Valves for Nuclear Power," Section B, Nuclear Class II Valves. Indication and control are available locally and on Panel No. 25 in the Control Room. Indication only is available on Panels 13 and 15. The valves fail closed with a loss of instrument air.

Steel Pipe Ducting--The steel pipe ductwork of the HCS is made of mild carbon steel with a six mil coat of Phenoline 368. It is designed for two psig positive pressure. The Reactor Building high pressure interlock which was provided to close AH-V3A and B at a setpoint pressure of 0.5 psig adequately protected the HCS steel pipe ductwork.

3.2.1.3 MHCS Protective Interlocks. The MHCS contained original, modified, and new interlocks to protect MHCS equipment and for rapid isolation on equipment failure or high radiation levels at the exhaust fan discharge. Table 7 lists all the protective interlocks pertinent to MHCS operation.

3.2.1.4 Instrumentation. Instrumentation for the MHCS included local pressure differential indication for the filter train's pre-filter, two HEPA filters, and activated carbon filter; a high filter train pressure differential alarm (Panel No. 25); temperature compensated measurement of MHCS exhaust flow; a gamma area monitor mounted in the MHCS filter plenum to measure the buildup of radioactive material on the filters; and an MHCS exhaust radiation monitor. The pressure differential indicators have been previously described.

Because the original HCS exhaust fan (150 cfm) was replaced with a 1000 cfm exhaust fan, the pressure transmitter (AH-DPT-5080) was replaced with a new transmitter, and it and the MHCS exhaust flow recorder were recalibrated to measure the increased MHCS flow. The exhaust flow recorder (AH-FR-5080) is located on Panel No. 25 in the Control Room.

A gamma area monitor was placed in the MHCS filter plenum upstream of the first HEPA filter to measure the buildup of radioactive material on the filters. The monitor had a local readout which was monitored frequently. If the contact radiation levels on the HEPA filter reached 1 rem/hr the venting procedure called for purge shutdown and changeout of the filter.

The exhaust radiation monitor, HPR-229, had a particulate, iodine, and gas channel. Since the expected release rate of Kr-85 exceeded the original instrument's monitoring range, HPR-229 was modified to include both a high and low range gas channel with sensitvities of 7.897 E2 cpm/ μCi/cc and 7.8 E7 cpm/ μCi/cc (Kr-85), respectively. The high range channel allowed monitoring Kr-85 levels up to 1000 μCi/cc. During venting, the setpoint for the alert alarm was 1014 cpm (1.28 μCi/cc) and for the high alarm was 1127 cpm (1.42 μCi/cc). The alert alarm setpoint was 90% of the high alarm setpoint and the high alarm setpoint activated at a Reactor Building concentration equivalent to the maximum previously measured Kr-85 concentration of 1.07 μCi/cc considering a meter error factor of 75%. Panel No. 12 in the Control Room contained the HPR-229 particulate, iodine, and low and high range gas channel

TABLE 7. MHCS PROTECTIVE INTERLOCKS

MHCS exhaust fan (AH-E-34) stops or cannot be started when:

- AH-V25 is closed (less than 20% open)
- High vacuum (greater than 15 inches W.G. vacuum) in the filter housing
- High alarm on HPR-229
- Loss of power to HPR-229
- Loss of instrument air to AH-V-36

AH-V7 closes upon:

- MHCS exhaust fan (AH-E-34) trip

AH-V36 closes upon:

- MHCS exhaust fan (AH-E-34) trip (30 second time delay)

AH-V3A & B close upon:

- Loss of power to AH-PS-5058 (Reactor Building high pressure switch)
- Reactor Building high pressure (greater than 0.5 psig)

instruments and the stripchart recorders (HP-UR-1907 Pens #13, 14, and 15 and HP-UR-3236 Pen #2, respectively) for the HPR-229 channels.

All instrumentation was checked and calibrated prior to the commencement of venting.

3.2.1.5 Tests and Inspections. To assure the operability of the MHCS prior to its use in venting the Reactor Building atmosphere, the MHCS was carefully tested in accordance with the "Functional Test Procedure for the Modified Hydrogen Control System" (SOP No. R-2-80-15). This test procedure was approved by NRC. The functional testing included demonstration of exhaust fan flow capacity, system trip interlocks, system alarms, and the operation of system valves. Also, in response to the earlier mentioned NRC concern with potential failure of the filter housing due to low pressure induced by the exhaust fan, the functional testing program included both dynamic and static tests to assure the filter housing could withstand up to 15 inch W.G. vacuum.

The HCS steel pipe ductwork was subjected to leak tests during manufacture, erection, and after assembly in the field. In order to ensure radiation would not be released from the ductwork during venting, a leak test of the ductwork downstream of the containment isolation valves and the filter housing was conducted prior to system operation. Testing was at 18 inches of water positive pressure and in accordance with ANSI N510, Section 6.3, 6.4, or 6.5. The indicated maximum leakage was less than six cfm/1000 ft^3 of system volume.

Filters and the filter housing were originally subjected to manufactures' performance and production tests as well as DOP and Freon II tests. Additionally, the filters of the MHCS were tested prior to the commencement of the Reactor Building venting. The carbon filter was subjected to a Freon II leakage test at 1000 cfm, the maximum flow expected in the system. The HEPA filters were subjected to an Efficiency Penetration Test (DOP). The HEPA filters were tested in accordance with ANSI N510-1975 and were verified to remove greater than or equal to 99.95% of the dioctyl-phtholate (DOP) while operating the system at a flow rate of 1000 cfm $\pm$ 10%.

The MHCS startup/test procedure was reviewed and approved by the NRC.

3.2.1.6 Operation of the MHCS. The MHCS was operated during the Reactor Building venting in accordance with Unit #2 Operating Procedure 2104-4.82. Controls for the MHCS were located on HVAC Panel No. 25 located in the Control Room. To start the system it was necessary first to open Reactor Building isolation valves AH-V3A and AH-V52. Throttle valve AH-V25 was then opened to at least 309° prior to starting the MHCS exhaust fan, AH-E-34. Upon starting the exhaust fan, AH-V36 was throttled to obtain the desired flowrate. To maintain Reactor Building pressure slightly below atmospheric, AH-V3B was opened and AH-V7 was opened and closed as necessary to replenish the exhausted air.

The system was shutdown by stopping AH-E-34 and closing AH-V25, AH-V52, AH-V3A and AH-V36 and AH-V3B and AH-V7.

3.2.2 Modified "B" Train of the Reactor Building Air Purge and Purification System

The MHCS was used initially during the venting of the Reactor Building because the high Kr-85 levels in the Reactor Building mandated low venting rates (see Section 3.3). However, as the Kr-85 concentration fell in the Reactor Building higher release rates were permitted. In order to complete the venting as quickly as possible the "B" train of the Reactor Building Air Purge and Purification System (RBPPS) was modified to allow venting at rates up to 18,500 cfm.

The RBPPS was originally designed to perform two functions: (1) provide clean heated air to the Reactor Building while purging clean filtered air to the environment and (2) recirculate and clean the Reactor Building air. A complete description of the Modified "B" train of the Reactor Building Purge and Purification System (MRBPS) used during the venting of Kr-85 from the Reactor Building is presented below.

3.2.2.1 System Description. Only the "B" train of the RBPPS was used for venting Kr-85 from the Reactor Building. The "B" train which required only minor modification consisted of a Reactor Building purge supply unit, a purge exhaust unit, and associated dampers, ductwork, and filters. The MRBPS is shown in Figure 1.

The MRBPS supply unit took suction from the intake tunnel. The supply unit consisted of a 25,000 cfm supply fan (AH-E-12B), a roll prefilter, a replaceable high efficiency filter (HEPA), and a multi-stage electric heater (not used during venting) all mounted in a steel cabinet. The supply unit inlet and outlet dampers (AH-D-5128B and AH-D-5128C) were interlocked to open with supply fan start. For the Reactor Building venting, the supply fan was disconnected so that upon placing the control switch for AH-E-12B to START, the dampers opened but the fan did not operate. The MRBPS supply line to the Reactor Building included an inner and outer isolation valve (AH-V2B and AH-V1B respectively). This supply unit allowed a flow path for purge makeup air to the Reactor Building under the reduced exhaust flow rate of MRBPS operation.

The MRBPS exhaust unit drew air from the Reactor Building through isolation valves AH-V3B (inner) and AH-V4B (outer) and discharged it to the station vent. The MRBPS exhaust unit consisted of a 25,000 cfm exhaust fan (AH-E-19B), roll prefilter, and two HEPA filters. The activated carbon filter normally part of the RBPPS filter train was not used during venting. The exhaust unit also included a manually adjusted filter housing inlet damper, a vortex damper integral with the exhaust fan (VD-5891B), an exhaust damper (AH-D-5129B), and a recirculation damper (AH-D-5129C). As part of the RBPPS, the vortex damper automatically maintained a negative pressure in the Reactor Building by throttling the exhaust flow. For Reactor Building venting, the vortex damper operation was changed from automatic to manual modulation. Hence, to control the purge flow between 1000 cfm and 7000 cfm the filter housing inlet damper was adjusted with the vortex damper "closed." Flow rates between 7000 cfm and about 18,500 cfm were obtained by adjusting the vortex damper with the filter housing inlet damper open. The 18,500 cfm flow rate maximum was due to the lack of supply fan operation, thereby increasing system resistance for the exhaust fan and lowering maximum flow from the 25,000 cfm design.

The RBPPS was designed to meet Class II seismic conditions, except for the exhaust filter train, the isolation valves, and the piping between the isolation valves which were designed to meet Class I seismic requirements.

3.2.2.2 Component Description.

MRBPS Supply Unit--The MRBPS supply unit is located in the Auxiliary Building at an elevation of 328 feet. The unit consists of a sheet metal cabinet containing the following equipment listed as they occur in the flow path:

(a) Prefilter AH-F-18B
(b) Electric duct heaters AH-C-47A-47H
(c) Air supply fan AH-E-12B

The cabinet is equipped with a walk-in door on both sides to permit easy maintenance.

As previously noted, the electric duct heaters were not utilized during venting and the supply fan AH-E-12A was made inoperable. Hence, neither are described here. The MRBPS air supply filter consists of an automatic roll media pre-filter and a HEPA cartridge filter (see Table 8).

The roll media pre-filter consists of a continuous, interlocked bonded fiberglass material having a nominal thickness of two inches when clean and does not compress more than one-quarter inch when subjected to air flow at 500 fpm. The media has a varying density in the direction of air flow enabling the dirt to penetrate the full depth of the media and eliminate the possibility of face loading. Each roll is reinforced for greater strength by steel wires firmly bonded to the exiting side of the media. The roll filter is reinforced on the air exiting side by string fiber mesh bonded to the roll of the media. The media is supported on the air entering and exiting sides by parallel steel wires running across the duct.

The roll filter media drive is actuated by a 0.5" differential pressure. The motor stops with a 0.45" differential. The drive assembly for the filter media consists of a 1/6 hp motor. The motor is equipped with thermal overload protection.

The HEPA filter consists of ultra fine bonded glass fiber housed in a corrosion resistant container.

Each filter has a local differential pressure indicator.

MRBPS Exhaust Unit--The MRBPS exhaust unit is located in the Auxiliary Building at an elevation of 328 feet. The exhaust unit consists of a sheet metal cabinet containing the following equipment listed as they occur in the flow path:

(a) Pre-Filter AH-F-19B
(b) HEPA Filter AH-F-20B

TABLE 8. MRBPS SUPPLY FILTER (AH-F-18B)

Pre-Filter	
Size	10'x6'8"
Capacity, cfm	25,000
Press. Drop (Clean), in. of H_2O	.16
Efficiency	85% (NBS Dust Spot Test)
Seismic Class	II
HEPA Cartridge Filter	
Size Per Cell, in.	24x24x21
Capacity, cfm	25,000
Press. Drop (Clean), in. of H_2O	0.5
Efficiency	93% (NBS Dust Spot Test)
Seismic Class	II

(c) Activated Carbon Filter AH-F-21B
(not used during Reactor Building venting)

(d) HEPA Filter AH-F-31B

(e) Air Exhaust Fan AH-E-19B

The cabinet is supplied with walk-in doors to permit easy maintenance.

Pre-Filter AH-F-19B--The pre-filter (Table 9) is an automatic renewable roll media filter similar to the MRBPS air supply pre-filter AH-F-18B described previously.

HEPA Filters, AH-F-20B and AH-F-31B--The HEPA filters (Table 9) are constructed of a dry fibrous high interception, sub-micron glass fiber which has an efficiency of 99.97% for particles larger than 0.3 microns. The filters conform to ORNL-NSIC-65. The filters are mounted in a steel frame and have aluminum separators. Each HEPA filter is fitted with a local differential pressure indicator.

MRBPS Exhaust Fan AH-E-19B--The MRBPS exhaust fan (Table 10) is a single width, single inlet, belt driven, centrifugal fan driven by a 60 hp motor. The fan is rated at 25,000 cfm at a static pressure of 11" H_2O. Control and indication are available locally and on Panel No. 25 located in the Control Room. To start the exhaust fan, either the discharge damper to the station vent or the recirculation damper must be open. The fan has a vortex damper which, as part of the RBPPS, throttled the exhaust flowrate to maintain a negative pressure in the Reactor Building. As previously discussed, the automatic control of the vortex damper was changed for venting to allow manual control.

Valves AH-V1B and AH-V4B--An air cylinder operated, 36" carbon steel butterfly valve with an ANSI rating of 100 psig and design temperature of 300°F is located in the RBPPS "B" train supply and exhaust lines, outside the Reactor Building at the 328 foot level of the Auxiliary Building. The valves are in full compliance with the "Draft ASME Code for Pumps and Valves for Nuclear Power," Section B, Nuclear Class II Valves. Control and indication is available locally and on Panel No. 25. Additional indication is available on Panels 13 and 15. The valves fail closed with a loss of instrument air.

TABLE 9. MRBPS EXHAUST FILTER TRAIN

Pre-Filter - AH-F-19B

Size, ft.	8x8
Capacity, cfm	25,000
Press. Drop (Clean), in. of H_2O	0.16
Efficiency	85% (NBS Dust Spot Test)
Seismic Class	I

HEPA Filters - AH-F-20B and -31B

No. of Cells Installed	40 (2 banks of 5x4)
Size, in.	24x24x11 1/2
Capacity Per Unit, cfm	1400
Press. Drop (Clean), in. of H_2O	1.2
Efficiency	99.97% for particles larger than 0.3 microns
Seismic Class	I

TABLE 10. MRBPS EXHAUST FAN

Identification	AH-E-19B
Type	Centrifugal
Rated capacity, cfm	25,000
Static pressure, in. of H_2O	11
Speed, rpm	1350
Fan Motor:	
Rated horsepower, hp	60
Speed, rpm	1800
Power requirements	460V/3ϕ/60Hz/68 amps full load current
Seismic Class	II
Other	Belt driven
	Variable inlet vanes

Valve AH-V2B--An air cylinder operated, 36" carbon steel butterfly valve with an ANSI rating of 100 psig and a design temperature of 300°F is located in the RBPPS "B" train supply line inside the Reactor Building at the 305 level. The valve is in full compliance with the "Draft ASME Code for Pumps and Valves for Nuclear Power," Section B, Nuclear Class II Valves. Indication and control is available locally and on Panel No. 25. Indication only is available on Panels 13 and 15. The valve fails closed with a loss of instrument air.

Valve AH-V3B--See discussion of this valve in Section 3.2.1.2.

MRBPS Supply Dampers D-5128B and D-5128C--An air operated, parallel blade damper is located in the inlet (D-5128B) and outlet (D-5128C) of the MRBPS supply unit. The inlet damper is the quick closing type which employs a return spring to achieve rapid closure. The dampers are interlocked with supply fan AH-E-12B to open when the fan starts and close when the fan stops. As previously noted, MRBPS supply fan AH-E-12B was disconnected but operation of the AH-E-12B control switch still opened and closed these two dampers as if the fan was operating. The dampers are also interlocked so that if they do not open within two seconds they will automatically reclose. With a loss of instrument air, D-5128B fails closed and D-5128C fails as is. When the control switch for AH-E-12B is moved to the START position, a red light will indicate the dampers are open.

MRBPS Exhaust Damper to the Station Vent D-5129B--An air operated, parallel blade damper is located in the outlet of the MRBPS exhaust duct. The damper is interlocked with the MRBPS exhaust fan so that either it or the recirculation damper D-5129C must be open to start the fan. The interlock which normally would close this damper upon a high radiation alarm from HPR-226 has been bypassed for the Reactor Building venting period (see Section 2.3). Control and indication are available locally and on Panel No. 25. The damper fails as is with a loss of instrument air.

MRBPS Recirculation Damper D-5129C--An air operated, parallel blade damper is located in the recirculation line which connects the MRBPS exhaust and supply lines. The damper automatically opens with a loss of instrument air. The interlock which normally would open this damper upon a high radiation alarm from HPR-226 has been bypassed for the Reactor Building venting period (see Section 2.3). Control and indication are provided locally and on Panel No. 25.

MRBPS Vortex Damper - VD-5891B--The vortex inlet damper to the MRBPS exhaust fan AH-E-19B was designed as part of the RBPP System to automatically maintain a negative pressure in the Reactor Building by throttling the exhaust flow. For the Reactor Building venting operation this vortex damper was modified from automatic to manual control. The controls for VD-5891B were positioned locally in the Auxiliary Building.

3.2.2.3 Protective Interlocks. The MRBPS included a number of protective interlocks. Table 11 summarizes the protective interlocks of importance to the MRBPS. Additional interlocks of the RBPPS existed and were functional during venting but were not important to the venting operation and therefore are not listed in Table 11. A description of the additional interlocks can be found in Reference 24.

The Reactor Building high pressure interlocks protected ductwork located outside the Reactor Building from rupture if there were a pressure rise in the Reactor Building and also prevented accidental radiation releases. The high radiation interlock from HPR-226 which normally would have closed the exhaust damper (D-5129B) and opened the recirculation damper (D-5129C) had been bypassed per a Technical Specification change described in Section 2.3.

3.2.2.4 Instrumentation. All filters in the MRBPS supply and exhaust system were supplied with local differential pressure indicators and all automatic roll filters had differential pressure switches to advance the media on a pre-set differential. Limit switches were provided to energize an alarm when the media was to be replaced.

The MRBPS also included instrumentation to monitor and record the supply (AH-FR-5075) and exhaust (AH-FR-5064) flows and to monitor the particulate, iodine, and gaseous radioactive material exhausted (HPR-226). As previously discussed, the interlock from HPR-226 that closed the MRBPS exhaust damper and opened the recirculation damper was bypassed per a Technical Specification change granted for venting (see Section 2.3). Panel No. 12 in the Control Room contains the HPR-226 particulate, iodine, and gas channel instruments and the stripchart recorder (HP-UR-2900 Pen #4, 5, and 6 respectively) for the HPR-226 channels.

All instrumentation was checked and calibrated prior to commencement of venting.

TABLE 11. PROTECTIVE INTERLOCKS FOR THE MRBPS

Valves AH-V1B, AH-V3B, and AH-V4B close or cannot be opened when:

- Reactor Building pressure exceeds 0.5 psig

MRBPS exhaust fan (AH-E-19B) stops or cannot be started when:

- Reactor Building pressure exceeds 0.5 psig
- Dampers D-5129B and C are both closed

MRBPS supply fan (AH-E-12B) trips upon:

- Failure of dampers D-5128B and C to open within two seconds following fan start

3.2.2.5 Tests and Inspections. The modified "B" train of the Reactor Building Air Purge and Purification System underwent functional testing prior to venting in accordance with functional test procedure SOP No. 2-R-80-38. This test procedure was approved by NRC. The functional testing included verification of system valve and damper operation, verification of system interlock operation, and determination of system exhaust flow control characteristics.

The MRBPS ductwork system was subjected to leak tests during manufacture, erection, and after assembly in the field. Filters and filter housings were subjected to manufactures performance and production tests as well as DOP tests. Prior to Reactor Building venting the entire MRBPS was visually inspected for potential leaks and suspected leaks were repaired. In addition, the HEPA filters in the exhaust unit were DOP tested in accordance with ANSI N510-1975.

3.2.2.6 Operation. The MRBPS was operated during the Reactor Building venting in accordance with Unit #2 Operating Procedure 2104-4.82. All controls for the system were located in the Control Room on the HVAC Panel No. 25 except for the vortex damper control which was located in the Auxiliary Building. System startup consisted of first adjusting the exhaust filter housing inlet damper and the vortex damper of AH-E-19B to a postion that would provide a flow rate less than that allowed. Discharge damper D-5129B was then opened followed by isolation valves AH-V3B and -V4B. The exhaust fan AH-E-19B was started and the flow rate adjusted using the exhaust filter housing inlet damper and the vortex damper. Reactor Building make-up air to maintain a slightly negative pressure was admitted by opening isolation valves AH-V2B and -V1B and then opening and closing, as required, D-5128B and C. (D-5128B and C were operated by the control switch for the RBPPS supply fan AH-E-12B.)

Shutdown of the system was accomplished by closing dampers D-5128B and C and valves AH-V1B and AH-V2B, stopping the exhaust fan AH-E-19B, closing valves AH-V4B and AH-V3B, and closing the vortex damper and damper D-5129B.

3.2.3 Reactor Building and Effluent Radiation Monitoring Equipment

The ability to accurately obtain the concentration of Kr-85 and other radionuclides in the Reactor Building atmosphere and to precisely measure the effluent radiation levels was essential to the venting program. Reactor Building direct air samples were taken either with the HPR-227 sampling system or from special equipment installed in containment penetration R-626. The Reactor Building air vented to the environment was monitored first by HPR-229 when using the MHCS or by HPR-226 when using the MRBPS. Then, following dilution with the exhaust from the Auxiliary Building and Fuel Handling Building ventilation systems it was again monitored in the station vent stack by HPR-219A.

The HPR-227 sampling system shown in Figure 2 can be used to fill a sample bomb for gas analysis, to perform a particulate or radio-iodine analysis by drawing Reactor Building air through a 100 millipore filter or a series of radio-iodine filters, or to perform tritium analysis using an installed bubbler. Separation of the different forms of iodine is accomplished based on the relative affinity of each iodine species for a specific filter medium in the series of iodine filters.

The HPR-227 sampling system normally takes samples from two points in the Reactor Building which are located approximately 10' 10" east and west of the north-south centerline of the Reactor Building dome (elevation 469'). The samples are transmitted through two lines running from the dome down and inside the Reactor Building and then through inner and outer isolation valves (AH-V106, AH-V103, and AH-V105, AH-V101) to the sample panel located in the Auxiliary Building. Exhausted Reactor Building air from the sample panel is discharged back to the Reactor Building through similar isolation valves (AH-V108, AH-V104, and AHV107, AH-V102). The sampling lines are designed to meet Seismic Class I requirements. Redundant inlet and discharge lines are provided to prevent a single active failure of any valve from impairing the function of the monitoring system. The isolation valves are all 1/2" solenoid operated stainless steel valves with a design pressure of 100 psig and a design temperature of 300°F. Control is provided locally and on Panel No. 25. Indication only is available on Panels 15 and 13.

The exact sampling location of HPR-227 was in doubt, however, due to the unknown position (open or closed) of the drain valves, AH-V182 and AH-V183, located on the sample lines inside the Reactor Building. If the drain valves were not closed, the HPR-227 samples would originate from both the dome area at the 469' elevation and the area near the drain valves which are located just inside the Reactor Building sampling line penetrations at approximately the 317' elevation. To alleviate this

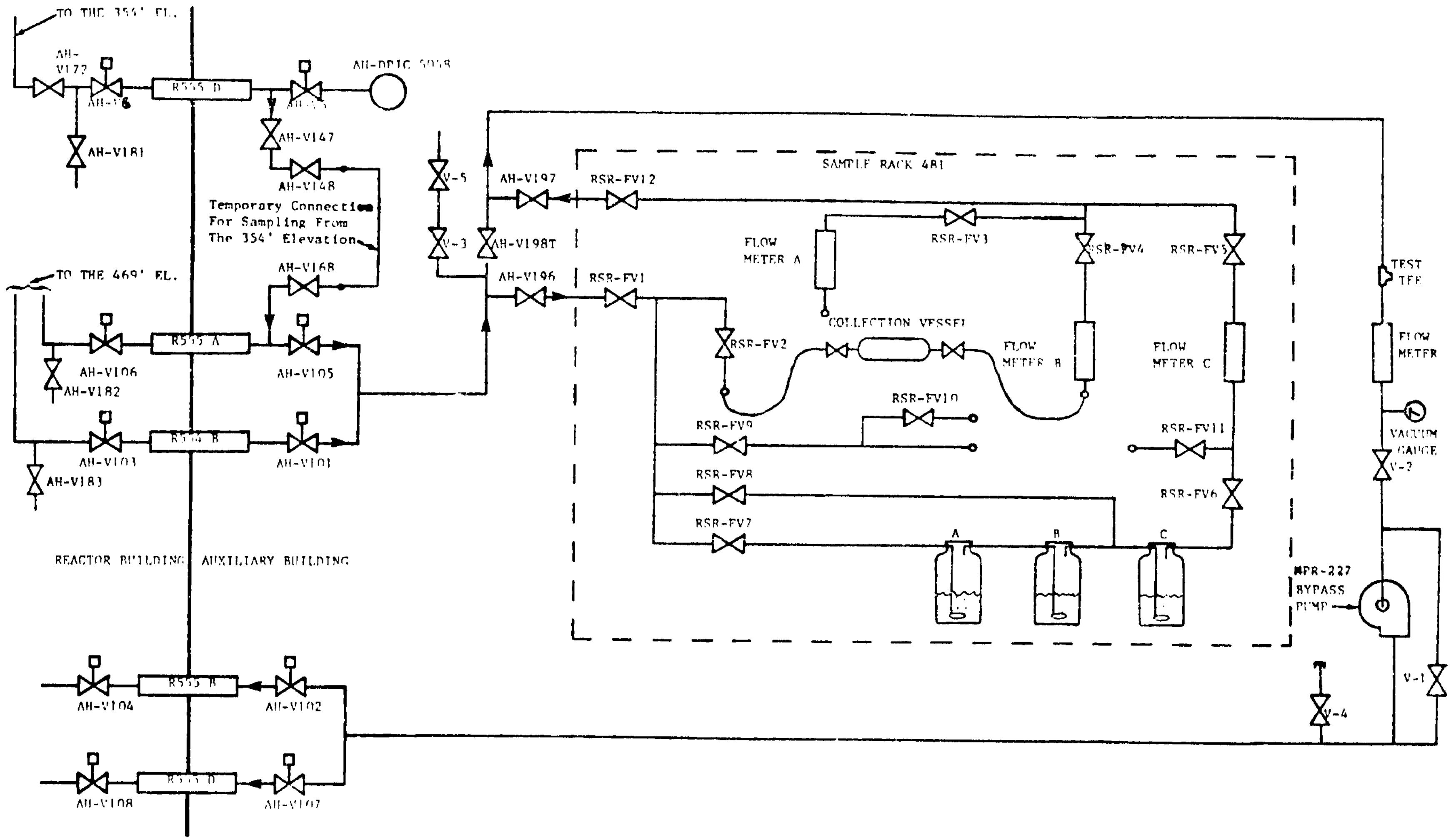

Figure 2. HPR-227 Bypass Sampling System Flow Diagram

uncertainty, a new sample line was established to the HPR-227 sample panel. This new sample line drew Reactor Building air samples at the 354' elevation by tying the HPR-227 sample panel to the Reactor Building pressure sensing line through its isolation test valves (AH-V147 and AH-V148) located between the sensing line isolation valves AH-V5 and AH-V6 (penetration R-562B). Reactor Building air samples were subsequently taken from both the original sample lines (469') and the new line (354').

Reactor Building gas samples are analyzed with a gas chromatograph to determine hydrogen content, and isotopic composition is determined with a gamma spectrum analyzer. The Kr-85 gas activity is determined by gamma spectroscopy techniques. Isotopic identification is made on the basis of the discrete energy levels at which gamma rays are absorbed in a germanium-lithium (Ge(Li)) detector. The particulate filters and radio-iodine filters are also analyzed using gamma spectroscopy.

GPU Technical Data Report #112[21] is a comprehensive discussion of the post-accident determination of the radioactive material content and chemical composition of the Reactor Building atmosphere and includes a more detailed description of the equipment and procedures which were utilized during venting to obtain Reactor Building air samples.

Reactor Building air samples were also obtained through containment penetration R-626 (Elevation 358'). Following the accident, a hole was drilled in this spare penetration and a glove box built enabling sampling of the Reactor Building atmosphere. Reactor Building gas, particulate, radio-iodine and tritium sampling similar to that from HPR-227 was conducted with the R-626 penetration glove box and sampling apparatus.

Effluent monitoring was done with HPR-229, HPR-226, and HPR-219A. HPR-229 is located immediately downstream of the MHCS exhaust fan, HPR-226 is immediately downstream of the MRBPS exhaust fan, and HPR-219A is the stack monitor. All three radiation monitors have a particulate, iodine, and gas channel. In addition, HPR-219A has the capability to take a tritium sample. HPR-229 and HPR-226 are further discussed in Sections 3.2.1.4 and 3.2.2.4 respectively. Additional information on their sensitivities and setpoints is available from Reference 25.

The stack monitor HPR-219A was the official instrument utilized to record the radiation releases during venting of the Reactor Building. HPR-219A is an Eberline radiation monitor and its readout is located on the turbine deck just outside the Control Room. The stack monitor was continuously monitored by a plant operator throughout the purge. Instantaneous, ten minute averages, hourly averages, and daily averages for beta particulate, iodine, and beta-gaseous activity exiting the station vent could be requested and printed out. Table 12 provides additional information on the HPR-219A channel sensitivities and the high and alert alarm setpoints. HPR-219A was checked and calibrated prior to the commencement of venting.

As discussed later in Section 3.5, false alarms on the HPR-219A particulate channel (response to Kr-85) led to the installation of two other particulate monitoring systems. The first was a simple particulate grab sample system whereby air was pulled from the stack sample line through a particulate filter which was replaced every 15 minutes during venting and immediately analyzed. The second system (HPR-219B) was more sophisticated and provided instantaneous (every 1000 seconds) readings of particulates being released. It consisted of a sodium iodide crystal detector which looked at a particulate filter through which air from the stack sample line was being pulled. This detector provided signals to a single channel analyzer where Kr-85 gamma activity was distinguished from other isotopes by looking only at Cs-137. This system had a lower limit of detectability of approximately 1.60 E-10 μCi/cc (or approximately 8.97 E-3 μCi/sec). Readout from this system was located on the turbine deck adjacent to the HPR-219A readout. The particulate release rate was based on the difference between the current and previous readings (1000 second intervals). A difference of 150 counts corresponded to a stack concentration of 5.8 E-10 μCi/cc or one-tenth the instantaneous particulate release rate Technical Specification limit, and was established as the "alarm" level.

3.2.4 Other Venting Support Systems, Equipment, and Instrumentation

3.2.4.1 Reactor Building Instrument Air Containment Isolation Valves AH-V60, -V61, -V62, and -V63. In order to operate the inner Reactor Building containment isolation valves AH-V2A, -2B, -3A, and -3B instrument air was required. Instrument air was supplied by two instrument air lines each of which contained an inner and outer isolation valve (AH-V61, -V63 and AH-V60, -V62). These valves are 1/2" stainless steel valves with a design pressure of 100 psig and a design temperature of 300°F.

TABLE 12. HPR-219A SENSITIVITY AND SETPOINTS[25]

HPR-219A Channel	Sensitivity	High Alarm	Alert Alarm
Particulate	1.34×10^{5} CPM/ μCi/cc (Sr-90)	2.0×10^{-3} μCi (80% of Tech. Spec. release rate limit of 0.3 μCi/Sec)	1.0×10^{-3} μCi (50% of high alarm setpoint)
Iodine	2.86×10^{4} CPM/ μCi/cc (I-131)	5.5×10^{-3} μCi (80% of Tech. Spec. release rate limit of 0.3 μCi/Sec)	2.7×10^{-3} μCi (50% of the high alarm setpoint)
Gas	2.6×10^{7} CPM/ μCi/cc	1.29×10^{-2} μCi/cc	1.16×10^{-2} μCi/cc

Control is provided locally and on Panel No. 25 in the Control Room. Indication is available locally and on Panels 13, 15, and 25. The operability of these valves was tested as part of the MHCS and MRBPS functional test procedures.

3.2.4.2 Reactor Building Pressure Instrumentation. Reactor Building pressure was closely monitored during venting to maintain the Reactor Building pressure between -0.5 and -0.1 inches of Hg. High Reactor Building pressure was used to automatically close containment isolation valves and shutdown equipment.

The Reactor Building pressure sensing line contains a solenoid operated one inch stainless steel valve with a design pressure of 100 psig and a design temperature of 300°F on both sides of Reactor Building penetration R-562B. These valves, AH-V6 and -V5, are located on the 305' level of the Reactor and Auxiliary Building respectively. Control is provided locally and on Panel No. 25 in the Control Room. Indication is available locally and on Panels 13, 15, and 25. The operability of these valves was checked as part of the MHCS and MRBPS functional test procedures.

Reactor Building pressure instrumentation of importance to the venting operation included AH-PS-5058 and a Reactor Building pressure-vacuum indicating gauge. AH-PS-5058 was interlocked to automatically close containment isolation valves AH-V1B, -V3A, -V3B, and -V4B and trip the MRBPS exhaust fan AH-E-19B upon sensing high Reactor Building pressure (setpoint 0.5 psi). The pressure-vacuum gauge located in the Auxiliary Building was used to monitor Reactor Building pressure. A closed circuit TV system was used to transmit a picture of the gauge to where the Control Room operators could easily read it. The TV monitor was originally located in the Service Building HVAC room underneath the Control Room but shortly following the start of venting it was moved to the new panel built adjacent to Panel No. 25 in the Control Room.

3.2.4.3 Area Radiation Monitoring. For the venting period, HPR-3236, the normal Reactor Building purge unit area radiation monitor, was moved and temporarily mounted near the MHCS filter train and exhaust fan to provide continuous monitoring for radiation leaks. HPR-3236 is a gamma/G-M monitor. It has local indication and alarm and also is indicated, recorded (HP-UR-1902, Pen #7), and alarmed on Control Room Panel No. 12. The alarm setpoint during the venting operation was set at 10 mr/hr. (See also Section 4.4.3 for a description of additional radiation monitoring conducted in the Auxiliary Building during venting).

3.2.4.4 Radiant Heaters for the MHCS Filter Train. One of the NRC concerns raised in their questions to Met-Ed/GPU on the proposed venting operation, was the possible degradation of HEPA filter efficiency due to moisture problems caused by the 100% relatively humidity which existed in the Reactor Building. An evaluation by Met-Ed showed that moisture formation on the filter media and the filter plenum and housing walls would only occur if the temperature of the surfaces was below the dew point of the air drawn through the plenum. To prevent any moisture formation, five infrared type radiant heaters were added along the outside of the filter plenum to elevate its surface temperature. The five heaters and their ability to heat the surface on the top of the filter housing were verified during functional testing of the MHCS (SOP No. R-2-80-15).

3.2.4.5 Station Vent Flow Instrumentation. To calculate the curies of radioactive material released during the venting of the Reactor Building, HPR-219A measurements were multiplied by the exhaust flow from the station vent. Station vent flow included not only the venting flow from the MHCS or the MRBPS but also the dilution flow fron the Auxiliary, Fuel Handling, and Service Building ventilation systems. The station vent flow was determined from a velocity probe and recorder. The recorder was located on the panel built adjacent to Panel No. 25 in the Control Room. Multiplication of the stack velocity by the cross-sectional area of the stack (70.85 ft^2) gave the flowrate in CFM. Prior to venting, the stack velocity measurement instrumentation was checked and calibrated.

3.2.4.6 Restoration of the Station Vent and Auxiliary and Fuel Handling Building Ventilation Systems. Prior to the venting of the Reactor Building, the plant vent stack cap was removed and use of the supplemental ventilation system atop the Auxiliary Building was discontinued. This was necessary to enable venting to the station vent and to restore the normal Auxiliary and Fuel Handling Building ventilation exhaust flows to provide the dilution required for venting. Met-Ed/GPU notified NRC of their intent to do this in a June 4, 1980 letter[14] which also included justification for this action. Part of the justification included how potential HEPA filter bypass paths in the Fuel Handling, Auxiliary, and Service Building ventilation systems had been prevented. Specifically, the activities which were completed prior to the removal from service of the supplemental ventilation/filtration system and the removal of the cap from the stack included:

- Inspection of all ductwork of the Auxiliary and Fuel Handling Building ventilation systems between the inlet of the exhaust fans and outlet of the filters for potential leak paths and sealing and testing of identified leaks.

- Re-balancing the Auxiliary and Fuel Handling Building ventilation systems to ensure their proper operation and correct flow rates.

- Re-testing the Auxiliary and Fuel Handling Building exhaust filters with DOP to the requirements of the Technical Specifications.

- Re-testing the Service Building HEPA filter, AH-F-28, with DOP.

3.3 Control

3.3.1 Venting Control

The venting of Kr-85 from the TMI-2 Reactor Building was carefully controlled to comply with the limits of 10 CFR Part 20, the objectives of 10 CFR Part 50, Appendix I, and the provisions of 40 CFR Part 190.0, to the extent they were applicable. The allowable off-site exposures resulting from venting were set by the revised Technical Specifications established by NRC's June 12, 1980 Order for Temporary Modification of License (see Section 2.3, Table 2). The revised Technical Specifications required that none of the following limits be exceeded for any of the 16 (22 1/2^{o}) sectors centered on the TMI-2 Reactor Building:

(a) 15 mrem skin dose
(b) 5 mrem total body dose
(c) 20% of the limits in (a) and (b) shall not be exceeded over any one hour period

These Technical Specification changes superceeded the previous instantaneous and quarterly average release rate limits for noble gases including Kr-85.

The above changes to the existing TMI-2 Technical Specifications were intended to provide flexibility in the venting process by expressing limits in terms of off-site doses rather than release rates. Thus, actual atmospheric dispersion conditions could be used to decrease the time required to complete the venting operation.

The previous instantaneous and quarterly average Technical Specification release rate limits for gross gaseous activity were developed for routine facility operation and phrased as limits on releases rather than limits on off-site doses (the effects of the releases). Therefore, compliance with 10 CFR Part 20 and Part 50 Appendix I, depended on on-site measurement of the amounts of material released and not on off-site dose measurements. However, the use of release limits instead of off-site dose limits dictated that an assumed conservative value for meteorological conditions (χ/Q) be chosen based on historical data, since meteorological conditions determine the off-site doses caused by the releases. It was this fixed worst case value of χ/Q that would have caused unnecessary delays had the Technical Specifications not been changed for the venting period since in real time, values both above and below the assumed χ/Q will occur. The revised Technical Specifications allowed the use of real-time meteorological data (χ/Q) to compute off-site doses. This permitted Met-Ed/GPU to take advantage of optimum dispersal conditions by increasing the release rate during favorable meteorology and complete the venting more expeditiously while still meeting the same 10 CFR Part 20 and Part 50, Appendix I, requirements which had also determined the previous release rate limits.

For the venting operation, a computer routine was developed which was capable of real-time calculation of the allowable venting flow rate. The allowed venting rate was based on the Kr-85 concentration in the Reactor Building, the current meteorological conditions (wind speed, wind direction and stability), and an allowed mrem/hr off-site exposure limit at or beyond the 600 m site exclusion radius. The Reactor Building Kr-85 concentration was input daily based on the latest Reactor Building air sample results. Meteorological data was input to the program from the TMI on-site meteorological tower every 15 minutes. The limiting mrem/hr skin and total body doses were also input parameters which were conservatively set initially at 0.1 and 0.03 mrem/hr respectively. In incremental steps during venting these limits were raised to 0.3 and 0.1 mrem/hr. Based on these inputs, the computer routine did atmospheric dispersion and radiological exposure calculations at least every hour to determine the maximum allowable venting rate. The atmospheric dispersion calculation was in accordance with Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from

Light Water-Cooled Reactors." Dose calculations were in accordance with Regulatory Guide 1.109, "Calculation of Annual Doses to Man From Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I."

A typical output of the computer routine is shown in Table 13. This computer printout was available in both the Control Room and the Environmental Assessment Command Center (EACC) and it provided the allowable purge flow rate in cubic feet per minute along with the key parameters upon which the calculation was based. Space was also provided on the printout for logging in the meteorological and venting conditions as indicated in the Control Room. In addition, the printout contained calculations of the percentage of MPC (PCT OF MPC), sum of Q/MPC (SUM OF Q/MPC), and percentage of the instantaneous technical specifications (PCT INST TECH SPEC) at the allowable purge flow rate. The latter two relate to the old technical specification limit of $Q_i/MPC_i \leq 1.5$ E5 m^3/sec. All three were useful as thumbrules/guideposts for monitoring releases. The allowable venting flow rate was normally calculated once per hour and the flow rate adjusted at that time. A calculation of the allowable venting flow rate could, however, be requested at any time.

The limiting skin and total body off-site dose rate levels input to the venting computer routine were set well below the NRC limit to assure that when calculations of the allowable venting flow rate, as described above, were made there would be adequate margin below the revised Technical Specifications. The computer routine also computed and accumulated beta skin and total body doses received in each of the 16 (22 1/2^{o}) sectors during the purge, based on meteorological data (15 minute intervals) and actual monitored release rates. These calculations allowed the identification of any sector which was approaching the (a) 15 mrem skin dose or (b) 5 mrem total body dose limits. If either of these limits was approached in any sector that sector was to be "blocked out" such that no venting could thereafter be allowed if meteorological conditions would have led to an additional incremental dose to that sector. As predicted by simulations made before the venting process using historical data, no sector reached the established levels at which it would have had to have been blocked out.

It should be pointed out that the computer routine for controlling the venting was based on the release of Kr-85. Release limits for all other radioisotopes were

TABLE 13. TYPICAL ALLOWABLE PURGE FLOW RATE COMPUTER PRINTOUT

```
THREE MILE ISLAND- RELEASE FLOW RATE COMPUTATION 06/28/80 15:16

DATA (YRMODYHRMN) -80 62815 0
SPEED (MPH)= 7.9
DIRECTION (WINDS FROM)= S
TEMPERATURE DIFF (F) = -1.4
STABILITY=B
                                       - - - - - - - - - - - - - - -
  INDICATED WIND SPEED =
                                       - - - - - - - - - - - - - - -
  INDICATED WIND DIRECTION =
                                   - - - - - - - - - - - - - - - - -
  INDICATED DELTA T =
                               - - - - - - - - - - - - - - - - - - -
  RB PRESSURE =
                               - - - - - - - - - - - - - - - - - - -
FOLLOWING VALUES ARE AFTER FLOW ADJUSTMENTS:
                                       - - - - - - - - - - - - - - -
  PURGE FLOW RATE =
                                   - - - - - - - - - - - - - - - - -
  HPR-219 KR85 CONC =
                                       - - - - - - - - - - - - - - -
  HPR-229 KR85 CONC =
                                       - - - - - - - - - - - - - - -

CONCENTRATION OF KR85   IN REACTOR BLDG (UCI/CC)= 1.02E 00

ATMOSPHERIC DISPERSION FACTOR            ENTRAINMENT COEF= .19
         X/Q(SEC/M3)            X/Q(SEC/M3)              X/Q(SEC/M3)
      603M = 1.1E-05         800M = 6.5E-06          1000M = 4.2E-06
     1500M = 1.6E-06        2000M = 8.2E-07          4000M = 1.6E-07
     8000M = 7.0E-08

DISTANCE GAMMA DOSE       BETA DOSE
     600.  9.11E-07 PEAK  1.00E-04 PEAK

PCT OF MPC= 218.2
SUM OF Q/MPC=  1.96E 05
PCT INST TECH SPEC= 130.7
ALLOWABLE ISOTOPIC RELEASE RATE (UCI/SEC)= 5.9E 04

    BASED ON BETA DOSE ---
* * * * * * * * * * * * * * * * * * * * * * * * * * * * * * * * * * * * * * *
*                                                                           *
*  ALLOWABLE PURGE FLOW RATE (CFM) =   122 (MIN=  116, MAX=   128)          *
*                                                                           *
*                                                                           *
* * * * * * * * * * * * * * * * * * * * * * * * * * * * * * * * * * * * * * *
```

shown in the Met-Ed/GPU Safety Analysis and Environmental Report, and in subsequent responses to NRC questions, to be met any time the Kr-85 limit was met, since Kr-85 was the dominant and controlling radioisotope.

In addition to the above computer routine which provided the primary control of venting, the Radiological Environmental Monitoring Program (REMP) conducted by Met-Ed/GPU (see Section 4.4.1) directly monitored off-site radiation levels. The monitoring teams were guided by near real-time estimates of plume location and intensity provided by the same computer system that provided venting rate calculations. The REMP supervisor had the authority based on the off-site measurement of radiation levels or other indications of adverse meteorological conditions to order a reduction or shutdown of venting. Other venting precautions and limitations dealing mainly with operability of equipment and instrumentation are described in Section 3.4.

3.3.2 Reactor Building Pressure Control

Reactor Building pressure was also carefully controlled during venting. Reactor Building pressure was maintained between -0.5 and -0.1 inches of Hg by opening and closing AH-V7 (AH-V3B open) when using the MHCS and by opening and closing D-5128B and C (AH-V1B and -V2B open) when using the MRBPS. Controls for all these valves and/or dampers and Reactor Building pressure indication were located on the HVAC Panel No. 25 or on the new panel adjacent to it in the Control Room. (During the initial venting period, Reactor Building pressure indication was located in the Service Building HVAC room located underneath the Control Room and two-way radios were used to establish communications between there and Panel No. 25.)

To ensure the Reactor Building could be maintained at a negative pressure by use of the Reactor Building air coolers in the event of venting shutdown, cooling water to the Reactor Building air coolers was secured during venting. The Reactor Building air cooling fans, however, were continuously operated during venting to insure good Reactor Building air mixing.

3.4 Operation

The entire Reactor Building venting operation was conducted in accordance with TMI-2 Operating Procedure 2104-4.82 "Reactor Building Atmosphere Cleanup Using the Modified Hydrogen Control System and the "B" Train of the Modified Reactor Building Purge System." As required by the NRC June 12, 1980 orders authorizing the venting and pursuant to Section 6.8.2 of proposed Appendix A to the Technical Specifications as made binding by the February 11, 1980 order of the Director, Office of Nuclear Reactor Regulation, this procedure was approved by NRC in addition to the normal Met-Ed approvals. The operating procedure included an introduction, references, limits and precautions, prerequisites, and step by step procedures for start-up and purging systems selection and start-up, normal operation, and temporary and final shutdown for both the MHCS and the MRBPS.

Section 3.3 has already discussed how the venting flow rate was controlled to keep the Kr-85 releases within the Technical Specification limits and how the Reactor Building pressure was controlled. In addition, Operating Procedure 2104-4.82 controlled the venting operation by requiring the shutdown of venting if any of the limitations listed in Table 14 occurred. Shutdown of venting was also required if:

(1) The particulate level of 6 E-10 μCi/cc gross beta-gamma was exceeded on the 15 minute samples from the bypass filter of HPR-219A.

(2) The particulate level of 6 E-10 μCi/cc gross beta-gamma was exceeded on the filter paper from HPR-219A which was exchanged and analyzed daily,

(3) The particulate level of 3 E-9 μCi/cc gross beta-gamma was exceeded on the HPR-229 filter paper which was exchanged and analyzed daily, or

(4) The number of counts in the Cs-137 channel of the real-time particulate monitor HPR-219B increased by 150 counts over the previous 1000 second reading (this is equivalent to a stack concentration of 5.8 E-10 μCi/cc or one-tenth of the instantaneous particulate release rate Technical Specifications limit).

Note: (1) and (3) were not limitations when HPR-219B, the real-time particulate monitor, was operating after the NRC approved its operation.

TABLE 14. LIMITS FOR VENTING OPERATION

Temporary shutdown of the MHCS was to be executed if any of the following limitations occurred.

(a) Inability to obtain purge data from the computer for more than one hour.

(b) Loss of HPR-219A.

(c) Valid High Activity Alarm on the HPR-219A gas channel.

(d) Valid High Radiation Alarm on HPR-3236.

(e) Valid High Activity Alarm on Local Portable Monitors.

(f) Off-site doses at limit as determined by the Site Environmental Impact Assessment Group.

(g) On-site doses at limit as determined by the Radiological Controls Department.

(h) Trip of Auxiliary and/or Fuel Handling Building HVAC Systems.

(i) Allowable Purge Flow Rate Less Than Minimum Flow.

(j) Filter Dose Rate greater than 1 R/hr.

(k) Loss of MHCS Flow Indication on AH-FR-5080.

(l) Inadvertant Closure of AH-V3A or AH-V52.

(m) High Filter Bank Differential Pressure Alarm on AH-PSA-5091.

(n) Loss of indication on stack flow velocity recorder.

Temporary shutdown of the MRBPS was to be executed if any of limitations a, b, c, d, e, f, g, h, k, or n for the MHCS above occurred or if any of the following limitatior occurred:

(1) Allowable purge flow rate less than the minimum achievable MRBPS exhaust flowrate.

(2) Loss of MRBPS flow indication on AH-FR-5064.

During the venting of the Reactor Building, there was one Control Room Operator (CRO) dedicated to the venting and who conducted all venting operations and one Assistant Operator (AO) dedicated to monitoring HPR-219A and later HPR-219B out on the Turbine deck (two-way radio communication between the AO on the turbine deck and the Control Room was provided). The overall supervision of the venting operation was by the shift foreman. Training classes were conducted for all operations personnel involved with the venting and these personnel were required to read and understand Operating Procedure 2104-4.82, the functional test procedures, and the flow print and to do a practical walk through of the system using the Operating Procedure as a guide. An oral examination was administered to insure these operating personnel had a satisfactory knowledge of the Reactor Building venting system.

At the end of each shift during venting an "R.B. Purge Operator Turnover Sheet" like the one shown in Table 15 was filled out. This sheet plus the operator and shift foreman logs and verbal communication assured smooth shift changes.

While venting was on-going, a shift engineer was also on continuous duty to monitor and help ensure the safe conduct of the venting. The shift engineer reported to the shift foreman and was delegated responsibility and authority including direction of the operators for controlling the venting. The shift engineer was also responsible for maintaining a complete record of the Kr-85 discharge (see Section 3.6.1.2) as well as a number of other duties including informing the environmental assessment command center (EACC) of each venting flow rate change and providing the EACC with the Reactor Building Kr-85 concentration, venting flow rate, and the Kr-85 release rate. NRC personnel were also on continuous 24-hour per day, seven days per week duty, overseeing the venting operation.

Because of the public interest and concern with the Reactor Building venting program, the venting operation included an emergency notification plan. As shown in Table 16, specific action levels were defined associated with possible venting occurrences. When an event of potential public interest or an unusual event occurred, specific personnel were notified so that they were cognizant of what had happened and so that the event could be properly reported to the public.

TABLE 15. R.B. PURGE OPERATOR TURNOVER SHEET

Date ________ CRO Assigned to Purge ________________

Shift ________ AO Assigned to Purge ________________

Purge Engineer ________________

Shift Foreman ________________

PURGE STATUS

Time ________

Present Flow Rate ________________ (SCFM)

Allowable Flow Rate ________________ (SCFM)

Maximum Flow Rate ________________ (SCFM)

Flow Rate Limited By ________________________________

RB Pressure ____________ (inches Hg)

VALVE STATUS (Indicate Position of below Valves)

No lights-open
Green light-closed
(Assumed Position)

AH-V5	________	AH-V3A	________
AH-V6	________	AH-V7	________
AH-V52	________	AH-V4B	________
AH-V25	________	AH-V1B	________
AH-V3B	________	AH-V2B	________

METHOD OF PARTICULATE MONITORING IN SERVICE (Circle one)

HP-R219B

15 Minute Sampling

PROBLEMS OR POSSIBLE PROBLEM AREAS

Off Going CRO ________________
Sign

On Coming CRO ________________
Sign

On Coming SF ________________
Sign

Attach completed form to S. F. Turnover

TABLE 16. REACTOR BUILDING VENTING EMERGENCY NOTIFICATION PLAN

Action Level	Action Level Characteristic
Event of Potential Public Interest	1. Commencement. 2. Restriction of purge in any given direction. 3. Purge system component failure. 4. Stack monitor (HPR-219A) alert alarm.
Unusual Event	1. Stack monitor (HPR-219A) high alarm. 2. Total integrated dose off-site (at any given location) greater than 0.2 mrem whole body or 10 mrem skin dose. 3. Purge flow rate higher than allowed by procedure. 4. Instantaneous whole body or skin dose reading off-site greater than 2.0 mrem/hour.

3.5 Venting Chronology

The chronology of events during the actual period of venting from June 28, 1980 to July 11, 1980 is presented in Table 17. Figure 3 provides a graphic picture of the venting chronology by showing the venting flow rates during this same period.

As can be seen from Table 17, the entire venting operation generally ran smoothly. The most serious problem occurred just after the initial commencement of venting. High alarms were received on the particulate channels of both HPR-219A and HPR-229 (set at 80% and 50% of the Technical Specifications limit of 0.3 μCi/sec respectively) after just four minutes of operation on June 28, 1980. Following shutdown of the system, the particulate filters from both these monitors were removed and analyzed but revealed no particulate activity. It was concluded, therefore, that the particulate detectors were responding to the Kr-85 in the system.

Later on that same day (June 28, 1980) venting resumed under test conditions to further evaluate system and associated monitor response with a very slow approach to the desired MHCS flow rate. During this testing period, additional filter samples were taken with subsequent analyses conducted to reaffirm that no particulate activity was present. HPR-219A was reprogrammed to subtract the gas channel reading from the particulate channel reading, but even with various correction factors, this proved to be unacceptable. Therefore, to monitor for particulate releases, two alternate systems were installed. One system was a bypass particulate sampler where particulate samples were taken every 15 minutes and analyzed immediately (see Section 3.2.3). This system was utilized until NRC approval of a second system which was a real-time particulate monitor system (HPR-219B) (see Section 3.2.3). The bypass particulate sampling system was thereafter used as a back-up to HPR-219B.

The only major occurrence was when the MRBPS was first used and krypton concentration levels in the Auxiliary Building rose, at one point, to approximately 186 times MPC levels. Subsequently, leaks were found and sealed in two ventilation system penthouse penetrations and in the doors leading into the penthouse. Auxiliary Building krypton concentration levels then dropped (see also Section 4.4.3).

TABLE 17. VENTING CHRONOLOGY

Date and Time		
June 28, 1980	0800	Commenced venting at 100 cfm using the MHCS
	0804	Temporary shutdown of venting and AH-V52 closure due to high particulate alarms on HPR-219A and HPR-229 Cause: Particulate detectors were responding to the Kr-85 being vented.
	0805	High alarm of HPR-219A gas channel Cause: HPR-219A dumped its computer programming and reverted to its cpm mode instead of μCi/cc mode on the gas channel.
	1700	Recommenced venting at low flow rates (15-89 cfm) for testing.
	1908	Temporary shutdown of test venting due to poor weather conditions (storm) for off-site environmental monitoring.
	2013	Recommenced test venting after storm had passed.
	2206	Temporary shutdown of venting - testing completed.
June 29, 1980	1400	Recommenced venting
	2139	Temporary shutdown of venting due to zero allowed flow rate from computer printout. Cause: Unknown
	2208	Recommenced venting

TABLE 17 (cont'd)

June 30, 1980	0152	MHCS exhaust fan AH-E-34 tripped - Temporary shutdown of venting. Cause: AH-E-34 accidently tripped by sheet metal workers touching fan shaft and causing motor overload.
	0350	Recommenced venting
	1235	Temporary shutdown of venting for filter change in radiation monitors. Note: HPR-219A must be shutdown to change filters.
	1258	Recommenced venting following completion of filter change.
	1700	Increased off-site beta dose limit from 0.10 to 0.20 mrem/hr skin dose.
July 1, 1980	1202	Temporary shutdown of venting for filter change on radiation monitors.
	1311	Recommenced venting following completion of filter change.
	1720	Increased off-site beta dose limit from 0.20 to 0.25 mrem/hr skin dose.

TABLE 17 (cont'd)

July 2, 1980	0013	Temporary shutdown of venting due to HPR-219A failure. <u>Cause</u>: I&C maintenance worker caused loss of HPR-219A readout while trying to repair the printout paper take-up.
		Temporary shutdown of venting was extended due to computer outage for maintenance.
		During this temporary shutdown installed air seal on shaft of MHCS exhaust fan AH-E-34 to reduce air inleakage.
	0400	HPR-219A returned to service.
	0532	Recommenced venting.
	1200	Temporary shutdown of venting for filter change on radiation monitors.
		Temporary shutdown of venting extended due to meteorological tower computer problems.
	1515	Recommenced venting following completion of filter change and meteorological tower computer repair.
July 3, 1980	0548	Temporary shutdown of venting due to loss of HPR-219A bypass particulate sample pump.
	0828	Recommenced venting following repair of HPR-219A by-pass particulate sample pump.

TABLE 17 (cont'd)

1100	HPR-219B, a new real-time particulate monitoring system, was approved for operation by NRC and replaced the requirement of 15-minute particulate filter samples from the HPR-219A bypass particulate sample system.
1202	Temporary shutdown of venting for filter change on radiation monitors.
1218	Recommenced venting following completion of filter change.
1310	Increased off-site beta dose limit from 0.25 to 0.30 mrem/hr skin dose.
1414	Temporary shutdown of venting due to Reactor Building pressure increase to -0.55 in. Hg. <u>Cause</u>: AH-V3B had been closed at the previous shutdown and had not been reopened thus making attempts to lower Reactor Building pressure by opening AH-V7 impossible.
1445	Recommenced venting.
1653	HPR-219B, the new real-time particulate monitor shutdown for recalibration - resumed 15-minute sampling with the HPR-219A bypass particulate sample system.
1951	HPR-219B returned to operation -- 15-minute particulate sampling ceased.

TABLE 17 (cont'd)

	2223	Reduced venting flow rate from 460 to 230 cfm on recommendation from REMP supervisor due to instantaneous beta readings of 1.5 mrem/hr in the vicinity of the TMI Observation Center.
July 4, 1980	0032	Temporary shutdown of venting for filter change on radiation monitors.
	0051	Recommenced venting following completion of filter change.
	2313	Temporary shutdown of venting due to poor meteorology.
		Filter change in radiation monitors also accomplished during this temporary shutdown.
July 5, 1980	0317	Recommenced venting.
	0945	Temporary shutdown of venting due to zero allowed flow rate from the computer printout. <u>Cause</u>: Beta dose limit was found to be zero.
	1040	Recommenced venting following resetting off-site beta dose limit to 0.3 mrem/hr skin dose.
July 6, 1980	0030	Temporary shutdown of venting for filter change on radiation monitors.
	0110	Recommenced venting following completion of filter change.

TABLE 17 (cont'd)

July 7, 1980	0030	Temporary shutdown of venting for filter change on radiation moni·ors.
	0105	Recommenced venting following completion of filter change.
July 8, 1980	0040	Temporary shutdown of venting for filter change on radiation monitors.
	0054	Recommenced venting following completion of filter change.
	0344	HPR-219B, the new real-time particulate monitor, readout/printout lost - resumed 15 minute sampling with the HPR-219A bypass particulate sample system.
	0624	Temporary shutdown of venting due to high alarm on HPR-229 Cause: HPR-229 failure suspected since no abnormal readings from HPR-219A.
	0648	HPR-219B repaired.
	0924	Reactor Building pressure at -0.1 in. Hg.
	0958	Reactor Building normal cooling water pump RB-P-1B, evaporative coolers RB-Z-1A and -1B, and two Reactor Building coolers were started to restore negative Reactor Building pressure.

TABLE 17 (cont'd)

	1224	Commenced venting using the MRBPS. Note: Krypton concentration levels in the Auxiliary Building rose following startup of the MRBPS. Levels reached approximately 186 times the MPC.
	1236	Secured RB-P-1B, RB-Z-1A, and -1B, and the two Reactor Building coolers.
	1345	Control Room ventilation placed on recirculation due to detection of krypton in the Control Room.
	1730	Started MHCS exhaust fan AH-E-34 with AH-V52 closed, AH-25 open, access door to MHCS filter housing nearest AH-V25 open, and AH-V36 full open to exhaust air from the 328' elevation of the Auxiliary Building to decrease the Kr-85 concentration caused by leakage from the MRBPS (per TCN 2-80-247 to Operating Procedure 2104-4.82)
(Approximate)	2200	MRBPS leaks (door to penthouse and two penthouse penetrations) identified and sealed.
July 9, 1980	0040	Temporary shutdown of venting (MRBPS) and shutdown of the MHCS which was being used to reduce Kr-85 levels in the Auxiliary Building.
	0058	Commenced venting using the MHCS. Note: Meteorological conditions would not allow use of the MRBPS.
	0430	Temporary shutdown of venting (MHCS) to allow change over to the MRBPS.

TABLE 17 (cont'd)

	0444	Commenced venting using the MRBPS. Note: The MHCS was run throughout most of the remainder of the venting period to reduce Kr-85 levels in the Auxiliary Building.
	0620	Temporary shutdown of venting (MRBPS) due to allowable flow rate less than 1000 cfm.
	0700	Commenced venting using the MRBPS.
July 10, 1980	0148	Temporary shutdown of venting (MRBPS) to allow Reactor Building air samples to be taken under non-venting conditions.
	1331	Recommenced venting using the MRBPS.
July 11, 1980	0100	Temporary shutdown of venting (MRBPS) for filter change on radiation monitors.
	0118	Recommenced venting using the MRBPS following completion of filter change.
	0933	Temporary shutdown of venting (MRBPS).
	1022	Venting ended, final shutdown of MHCS and MRBPS.

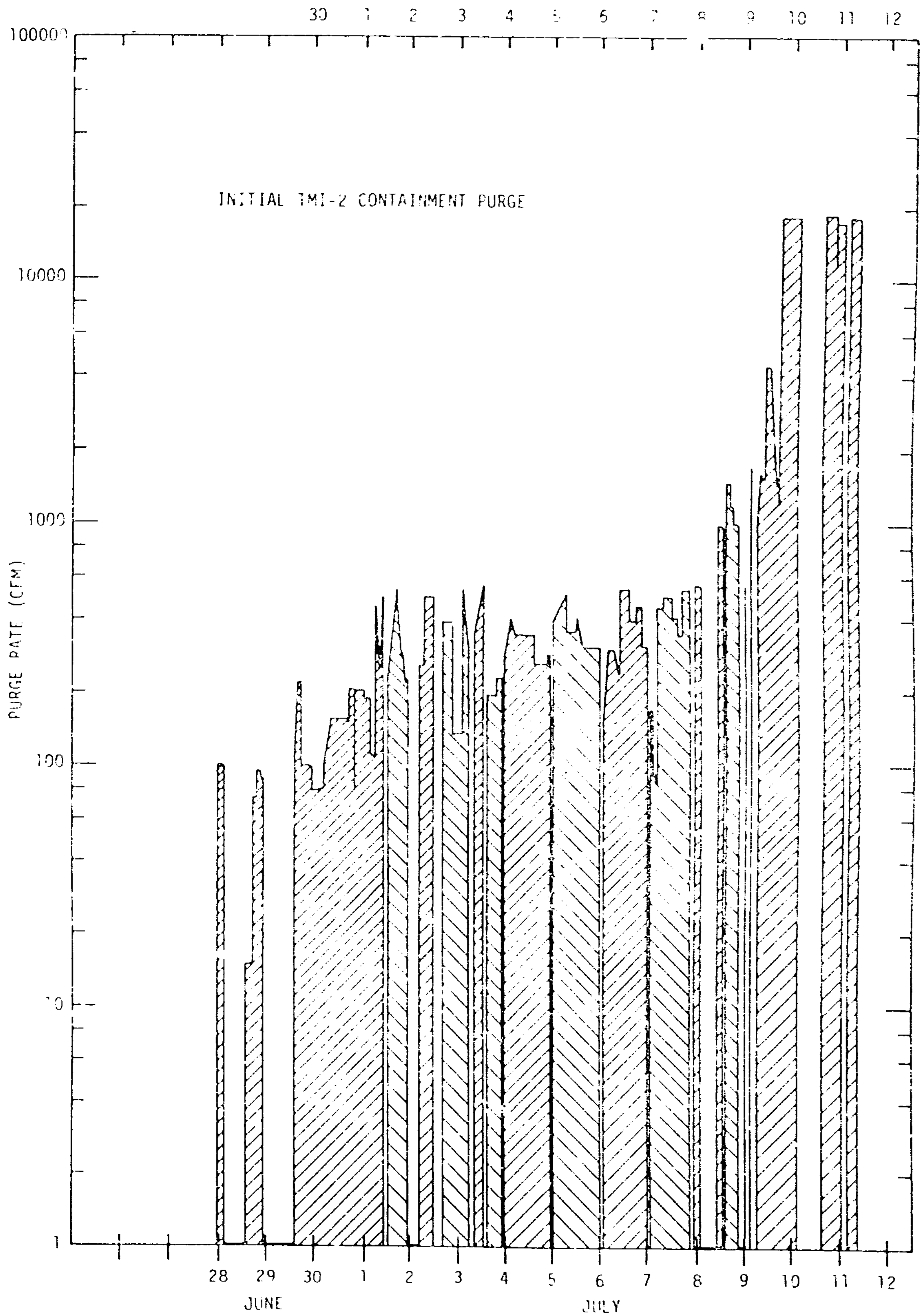

Figure 3. Histogram of Venting Rates June 28 to July 11, 1980

All of the other venting problems were relatively minor in nature (computer problems, spurious instrument alarms, etc.). Problems involving discrepancies in calculated versus measured curies of Kr-85 vented, and measured curie releases versus the original estimated Reactor Building inventory, are discussed in Section 3.7.

3.6 Data Summary and Results

3.6.1 Reactor Building Venting Records

3.6.1.1 Operator Logs. Throughout the entire venting period hourly logs of important venting parameters were kept by the Control Room Operator (CRO) responsible for the venting operation. The CRO on the hour recorded in the Recovery Station Daily Purge Log Sheet the following information:

- AH-V-36 position (MHCS flow control valve)
- Stack Flow in FPM and CFM
- MHCS flow
- MRBPS flow
- Delta temperature, atmospheric temperature, wind speed, and wind direction from the TMI meteorological station
- Reactor Building pressure
- Radiation level near the MHCS and MRBPS (HPR-3236)
- Radiation levels (particulate, iodine, and gas) in the MHCS or MRBPS exhaust (HPR-229 and HPR-226 respectively) dependent on which system was operating.
- Whether or not the interlocks associated with HPR-229 and HPR-226 were in defeat.
- Radiation levels (particulate, iodine, and gas) in the stack exhaust.
- Radiation level on the MHCS filters (read only once per shift).
- Differential pressure across the MHCS or MRBPS filters (read only once per shift).
- Allowed venting flow rate (from computer printout).

3.6.1.2 Shift Engineer Logs. The shift engineer also kept a record of venting parameters. Table 18 provides a description of the entries made in the "Shift Engineers' Purge Discharge Record" on the hour or whenever the venting flow rate was adjusted. This log was primarily aimed at monitoring the Kr-85 release rate and total hourly curies of Kr-85 released. Columns (13) to (19) were supposed to have been used to calculate the Kr-85 release rate based on MHCS flow and HPR-229 for comparison with the release rate calculated from the stack flow and HPR-219A. Because of early differences between the two, columns (14) through (19) were omitted during most of the venting. (Section 3.7 provides further discussion of the reasons for the difference in calculated release rates using these two different methods.)

3.6.1.3 Computer Printout. The third major type of venting record kept was the computer printout sheets that gave the maximum allowable venting rates (see Section 3.3). A copy of a typical printout is shown in Table 13. These printouts were provided automatically once per hour or more often if requested. They contained the allowable venting rate and the basic parameters from which it was determined. The printout also had space to enter at the time of the printout:

- The present wind speed, wind direction, and delta temperature from the TMI meteorological tower indicated in the Control Room so a quick comparison could be made with the meteorological parameters used to calculate the allowable flow rate,
- the Reactor Building pressure, and
- after flow adjustments, the new venting flow rate, HPR-219A Kr-85 concentration, and the HPR-229 (or HPR-226) Kr-85 concentration

3.6.2 Reactor Building Atmosphere Sampling

As previously discussed in Section 3.1, direct air sampling had fairly well established the Reactor Building atmosphere concentrations for the various radioisotopes. Kr-85 was by far the dominant isotope and was the only radioisotope of importance as far as the venting program was concerned. Before the commencement of venting, however, a final baseline sample was taken for noble gases, iodine, tritium, and particulates. The results are presented in Tables 19 and 20.

TABLE 18. DESCRIPTION OF SHIFT ENGINEER'S PURGE DISCHARGE RECORD

COLUMN NUMBER	DATA SYMBOL	DESCRIPTION OF CALCULATIONS AND DATA TO BE ENTERED INTO THE RECORD
(1)	DATE	Enter the date.
(2)	T_1	Enter the time in hr:min at the start of each purge period after the flow adjustment.
(3)	T_2	Enter the time in hr:min at the end of each purge period.
(4)	ΔT	Enter the duration of each purge period in minutes at the end of each purge period. The duration is given by: $\Delta T = T_2 - T_1$
(5)	V_{STK}	Enter the stack flow velocity in fpm at the start of each purge period. The stack flow velocity is read on the recorder to the left of Panel 25. Note: There are 35 fpm/chart division.
(6)	F_{STK}	Enter in cfm the plant stack flow rate which is computed from the stack flow velocity as follows: $F_{STK} = A_{STK} V_{STK}$ where: V_{STK} = stack flow velocity in fpm from (5); A_{STK} = cross sectional area of the stack in ft^2 $= \frac{\pi}{4} D^2_{STK}$ $= \frac{\pi}{4} (9.5)^2$ $= 70.88\ ft^2$ Therefore: $F_{STK} = 70.88\ V_{STK}$

TABLE 18 (cont'd)

COLUMN NUMBER	DATA SYMBOL	DESCRIPTION OF CALCULATIONS AND DATA TO BE ENTERED INTO THE RECORD
(7)	C_{STK}	Enter the stack concentration of Kr-85 in μCi/cc at time T_I. This concentration is read directly from the stack monitor HPR-219A.
(8)	Q_{STK}	Compute and enter in μCi/sec the plant release rate at time T_I and report the value to the REM supervisor. The release rate is computed as follows: $Q_{STK} = F_{STK} \times C_{STK} \times K$ where: Q_{STK} = μCi/sec F_{STK} = cfm from (6) C_{STK} = μCi/cc from (7) K = units conversion factor $K = \frac{1\text{ min}}{60\text{ sec}} \times \frac{28317\text{ cm}^3}{\text{ft}^3}$ $K = 472$ Therefore: $Q_{STK} = F_{STK} \times C_{STK} \times 472$
(9)	$\bar{V}_{STK}$	Enter in fpm the time average of the stack flow velocity over the duration of the purge period, ΔT. The average velocity is read on the recorder to the left of Panel 25.
(10)	$\bar{F}_{STK}$	Enter in cfm the time average of the stack flow rate over the duration of the purge period, ΔT. The average flow rate is computed from the average stack flow velocity in the same manner as (6), i.e.: $\bar{F}_{STK} = 70.88\ \bar{V}_{STK}$ Note: The plant stack flow rate, $\bar{F}_{STK}$, may be computed directly from the number of divisions, D, read from the stack velocity recorder as follows: $\bar{F}_{STK} = 70.88 \times 35 \times D = 2481 \times D$

TABLE 18 (cont'd)

COLUMN NUMBER	DATA SYMBOL	DESCRIPTION OF CALCULATIONS AND DATA TO BE ENTERED INTO THE RECORD
(11)	$\bar{C}_{STK}$	Enter in μCi/cc the time average of the stack concentration of Kr-85 over the duration of the purge period, ΔT. The average concentration is read directly from the stack monitor HPR-219A.
(12)	$\bar{Q}_{STK}$	Compute and enter in μCi/sec the average plant release rate over the duration of the purge period, ΔT. The average release rate is computed in the same manner as (8), i.e.: $\bar{Q}_{STK} = \bar{F}_{STK} \times \bar{C}_{STK} \times 472$
(13)	$\bar{F}_{MHCS}$	Enter in cfm the time average of the MHCS exhaust flow rate over the duration of the purge period, ΔT. The average flow is read from the strip chart of AH-FR-5080 on Panel 25.
(14)	$\overline{CPM}_H$	Enter in cpm the time average of the high range rate meter for HPR-229X. The average count rate is read from the strip chart of HP-UR-3236 (point #1) on Panel 12.
(15)	$\bar{C}_{MHCS}$	Enter in μCi/cc the time average of the MHCS exhaust concentration of Kr-85 over the duration of the purge period, ΔT. The average concentration is computed by dividing $\overline{CPM}_H$ from (14) by the high range monitor sensitivity, i.e.: $\bar{C}_{MHCS} = \frac{\overline{CPM}_H}{879.6}$
(16)	$\bar{Q}_{MHCS}$	Compute and enter in μCi/sec the average MHCS release rate over the duration of the purge period, ΔT. The average release rate is computed in the same manner as (8), i.e.: $\bar{Q}_{MHCS} = \bar{F}_{MHCS} \times \bar{C}_{MHCS} \times 472$

TABLE 18 (cont'd)

COLUMN NUMBER	DATA SYMBOL	DESCRIPTION OF CALCULATIONS AND DATA TO BE ENTERED INTO THE RECORD
(17)	$\overline{CPM}_L$	Enter in cpm the time average of the low range rate meter for HPR-229X. The average count rate is read from the strip chart of HP-UR-1907, (point #15) on Panel 12.
(18)	$\bar{C}_{MHCS}$	Enter in μCi/cc the time average of the MHCS exhaust concentration of Kr-85 over the duration of the purge period, ΔT. The average concentration is computed by dividing $\overline{CPM}_L$ from (17) by the low range monitor sensitivity, i.e.: $\bar{C}_{MHCS} = \dfrac{\overline{CPM}_L}{7.8\text{ E}7}$
(19)	$\bar{Q}_{MHCS}$	Compute and enter in μCi/sec the average MHCS release rate over the duration of the purge period, ΔT. The average release rate is computed in the same manner as (8), i.e.: $\bar{Q}_{MHCS} = \bar{F}_{MHCS} \times \bar{C}_{MHCS} \times 472$
(20)	A	Compute and enter in Ci the total activity released over the duration of the purge period, ΔT. The total is computed as follows: $A = \Delta T \times \bar{Q}_{STK}$

TABLE 19. HPR-227 PARTICULATE, IODINE, AND TRITIUM SAMPLING RESULTS

Date and Time	Sample No. (Type)	Sample Location	Results[a]	
6/27/80 - 0940	43561 (Gas)	354' Elevation	H_2 = 0.2% O_2 = 17.3% N_2 = 82.5%	
6/27/80 - 1220	43570 (Tritium)	354' Elevation	H-3 = (1) 1.42E-5 μCi/cc (2) 1.4E-5 μCi/cc	
6/27/80 - 1050	43571 (Particulate)	354' Elevation	Gross Beta-Gamma -	1.365E-10 μCi/cc
			Sr/Y-90 -	< 8.0E-11
			Gross Alpha -	5.41E-12 μCi/cc
			Gamma Analysis -	No Detectable Isotopes *Others < E-11
6/27/80 - 1000	43572 (Particulate)	354' Elevation	Gamma Analysis -	Kr-85 < E-8 I-131 < E-11 *Others < E-10

* Others = Cs-134, Cs-137, Co-58, and Co-60

[a] Less than indicates these nuclides are below the listed instrumentation sensitivity for these nuclides (LLD)

TABLE 19 (cont'd)

Date and Time	Sample No. (Type)	Sample Location	Results[a]	
6/27/80 - 1050	43573 (Charcoal)	354' Elevation	Gamma Analysis-	Kr-85 = 2.1E-6 μCi/cc I-131 < E-9 *Others < E-10
6/27/80 - 1640	43588 (Particulate)	---	Gross Beta-Gamma -	1.92E-9 μCi/cc
			Sr/Y-90 -	< 2.6 E-10
			Gross alpha -	4.826E-12 μCi/cc
			Gamma Analysis -	*Others < E-10
6/27/80 - 1554	43589 (Charcoal)	---	Gamma Analysis -	Kr-85 = 6.18E-6 μCi/cc *Others < E-10
6/27/80 - 1502	43597 (Tritium)	469' Elevation	H-3 = 1.3E-5 μCi/cc	

* Others = Cs-134, Cs-137, Co-58, and Co-60

[a] Less than indicates these nuclides are below the listed instrumentation sensitivity for these nuclides (LLD)

TABLE 19 (cont'd)

Date and Time	Sample No. (Type)	Sample Location	Results[a]	
7/4/80 - 1200	44542 (Particulate)	469' Elevation	Gross Beta-Gamma - Sr/Y-90 - Gross alpha - Gamma Analysis-	3.30E-11 μCi/cc < 2.25E-10 8.95E-12 μCi/ml I-131 < E-11 *Others < E-10
7/4/80 - 1140	44543 (Tritium)	469' Elevation	H-3 = (1) 9.8E-6 μCi/cc (2) 9.4E-6 μCi/cc	
7/7/80 - 1100	44680 (Charcoal)	---	Gamma Analysis -	ALL < E-10 (both sides)
7/11/80 - 1250	44189 (Tritium)	469' Elevation	H-3 = (1) 8.0E-6 μCi/cc (2) 8.1E-6 μCi/cc	

* Others = Cs-134, Cs-137, Co-58, and Co-60

[a] Less than indicates these nuclides are below the listed instrumentation sensitivity for these nuclides (LLD)

TABLE 19 (cont'd)

Date and Time	Sample No. (Type)	Sample Location	Results[a]		
7/11/80 -	45190 (Particulate and charcoal)	469' Elevation	Charcoal:	Gamma Analysis -	I-131 < E-10
					*Others < E-9
			Particulate:	Gross Beta-Gamma -	3.87E-10 μCi/cc
				Sr/Y-90 -	< 1.8E-10
				Gross alpha -	2.14E-11 μCi/cc
				Gamma Analysis-	I-131 < E-9
					*Others < E-9
7/11/80 - 1500	45199 (Particulate and charcoal)	354' Elevation	Charcoal:	Gamma Analysis -	*Others < E-9
					Ag-110M < E-9
					Mn-54 < E-9
			Particulate:	Gross Beta-Gamma -	2.65 E-10 μCi/ml
				Gross alpha -	2.03E-11 μCi/ml
				Gamma Analysis -	I-131 < E-11
					Cs-137 = 2.010E-10 μCi/cc
					*Others < E-10
					Mn-54 < E-11

* Others = Cs-134, Cs-137, Co-58, and Co-60

[a] Less than indicates these nuclides are below the listed instrumentation sensitivity for these nuclides (LLD)

TABLE 19 (cont'd)

Date and Time	Sample No. (Type)	Sample Location	Results[a]
7/11/80 - 1500	45200 (Tritium)	354' Elevation	H-3 = 4.8E-6 μCi/cc

* Others = Cs-134, Cs-137, Co-58, and Co-60

[a] Less than indicates these nuclides are below the listed instrumentation sensitivity for these nuclides (LLD)

TABLE 20. REACTOR BUILDING Kr-85 SAMPLING RESULTS

Sample Date and Time	Sample No.	Type of Sample	Sample Location	Results (μCi/cc)
6/27/80 - 0955	43560	S[a]	354' Elevation	1.02 $\pm$ 0.0044
6/27/80 - 1505	43587	S	469' Elevation	0.96 $\pm$ 0.0042
6/29/80 - 0853	43754	S	469' Elevation	0.95 $\pm$ 0.00414
6/29/80 - 0912	43755	S	469' Elevation	0.88 $\pm$ 0.00404
6/30/80 - 0920	43880	S	469' Elevation	0.91 $\pm$ 0.0043
6/30/80 - 0930	43881	S	469' Elevation	0.92 $\pm$ 0.0043
6/30/80 - 1040	43882	S	354' Elevation	1.01 $\pm$ 0.0042
6/30/80 - 1047	43883	S	354' Elevation	0.998 $\pm$ 0.0043
7/1/80 - 0906	44082	S	469' Elevation	0.884 $\pm$ 0.004
7/1/80 - 0915	44083	S	469' Elevation	0.89 $\pm$ 0.004
7/2/80 - 1042	44301	S	469' Elevation	0.72 $\pm$ 0.0036
7/2/80 - 1144	44302	S	469' Elevation	0.71 $\pm$ 0.0036
7/3/80 - 0900	44429	S	469' Elevation	0.61 $\pm$ 0.0033
7/3/80 - 1007	44428	S	469' Elevation	0.61 $\pm$ 0.0033
7/4/80 - 1020	44541	S	469' Elevation	0.461 $\pm$ 0.0029
7/4/80 - 1131	44544	S	469' Elevation	0.468 $\pm$ 0.0029
7/5/80 - 1013	44587	S	469' Elevation	0.372 $\pm$ 0.0026
7/5/80 - 1116	44588	S	469' Elevation	0.358 $\pm$ 0.0025
7/5/80 - 1127	44589	M[b]	469' Elevation	0.285 $\pm$ 0.000196
7/5/80 - 1442	44590	M	354' Elevation	0.284 $\pm$ 0.00053

[a] S = Sausage Sample, 30 ml

[b] M = Marinelli Sample, 1640 ml

TABLE 20 (cont'd)

Sample Date and Time	Sample No.	Type of Sample	Sample Location	Results (μCi/cc)
7/5/80 - 1555	44586	S	354' Elevation	0.328 ± 0.0023
7/5/80 - 1555	44585	S	354' Elevation	0.342 ± 0.0025
7/6/80 - 1108	44641	S	469' Elevation	0.259 ± 0.00216
7/6/80 - 1217	44642	S	469' Elevation	0.263 ± 0.0022
7/6/80 - 1340	44647	M	R626	0.216 ± 0.00046
7/7/80 - 0045	44682	S	354' Elevation	0.215 ± 0.002
7/7/80 - 0045	44683	S	354' Elevation	0.203 ± 0.002
7/8/80 - 1035	44835	S	469' Elevation	0.134 ± 0.0016
7/8/80 - 1143	44836	S	469' Elevation	0.135 ± 0.0016
7/9/80 - 1146	44940	S	354' Elevation	0.0072 ± 0.00037
7/9/80 - 1252	44941	S	354' Elevation	0.0083 ± 0.00041
7/9/80 - 1305	44942	M	354' Elevation	0.00705 ± 0.000085
7/9/80 - 1327	44945	M	469' Elevation	0.0076 ± 0.000086
7/9/80 - 1428	44943	S	469' Elevation	0.0094 ± 0.000556
7/9/80 - 1433	44944	S	469' Elevation	0.00804 ± 0.00039
7/10/80 - 0330	45026	M	354' Elevation	0.000173
7/10/80 - 0335	45021	M	354' Elevation	0.00017 ± 0.000013
7/10/80 - 0400	45028	M	354' Elevation	0.00015 ± 0.000011
7/10/80 - 0600	45034	M	354' Elevation	0.000186 ± 0.000013
7/10/80 - 0610	45035	M	354' Elevation	0.000189 ± 0.000014
7/10/80 - 0615	45036	M	354' Elevation	0.000199 ± 0.0000138
7/10/80 - 0732	45037	M	354' Elevation	0.000178 ± 0.0000133

TABLE 20 (cont'd)

Sample Date and Time	Sample No.	Type of Sample	Sample Location	Results (µCi/cc)
7/11/80 - 1129	45191	M	469' Elevation	0.000035 ± 5.1E-6
7/11/80 - 1232	45192	M	469' Elevation	0.0000356 ± 6.4E-6
7/11/80 - 1522	45196	M	354' Elevation	0.000058 ± 7.3E-6
7/11/80 - 1534	45197	M	354' Elevation	0.000064 ± 7.2E-6

During the venting period, at least one daily sample of the Reactor Building atmosphere was taken and analyzed for noble gases. All samples were taken at sample panel HPR-227 (469' or 354' elevation) except for one taken at containment penetration R-626. These results were used to monitor the progress of venting and also to update the computer routine calculating the allowable venting flow rate (see Section 3.3). Table 20 gives the results of all noble gas samples taken from the Reactor Building during venting. Figure 4 is a plot of the Kr-85 concentration in the Reactor Building based on these results during the venting period. As can be seen in Table 20, the samples drawn from the 354' and 469' elevations gave similar results verifying adequate mixing of the Reactor Building atmosphere.

Weekly, during the venting period, a Reactor Building particulate sample was taken with HPR-227 and analyzed for isotopic content and gross beta-gamma activity. Since the venting only lasted 14 days, only one sample was taken. The results of this sample are presented in Table 19. In addition, a tritium sample and an iodine sample were taken midway during venting and their results are shown in Table 19. Immediately following the completion of venting, particulates, iodine, and tritium samples were again taken. Table 19 also presents the results of these samples.

Starting on July 4, 1980, particulate and iodine samples were also taken using specialized sampling equipment installed through containment penetration R-626 (see Section 3.2.3 and Reference 26). Four continuous samples were taken over the periods of July 4-6, July 6-8, July 8-9, and July 10-12. The analysis results of these samples are provided in Table 21.

3.6.3 Effluent Radiation Monitoring

As discussed in Section 3.2.3, releases of radioactive material during venting were officially monitored by the station vent radiation monitor HPR-219A. However, because of the early problem with the HPR-219A particulate channel, two alternate particulate monitoring systems, a bypass particulate filter which was changed every 15 minutes and immediately analyzed and a real-time particulate monitoring system (HPR-219B), were used during most of the venting period to monitor station vent particulate releases. The exhausts from the MHCS and MRBPS were also directly monitored by HPR-229 and HPR-226, respectively. The results of data gathered from all these monitors is presented in the following subsections.

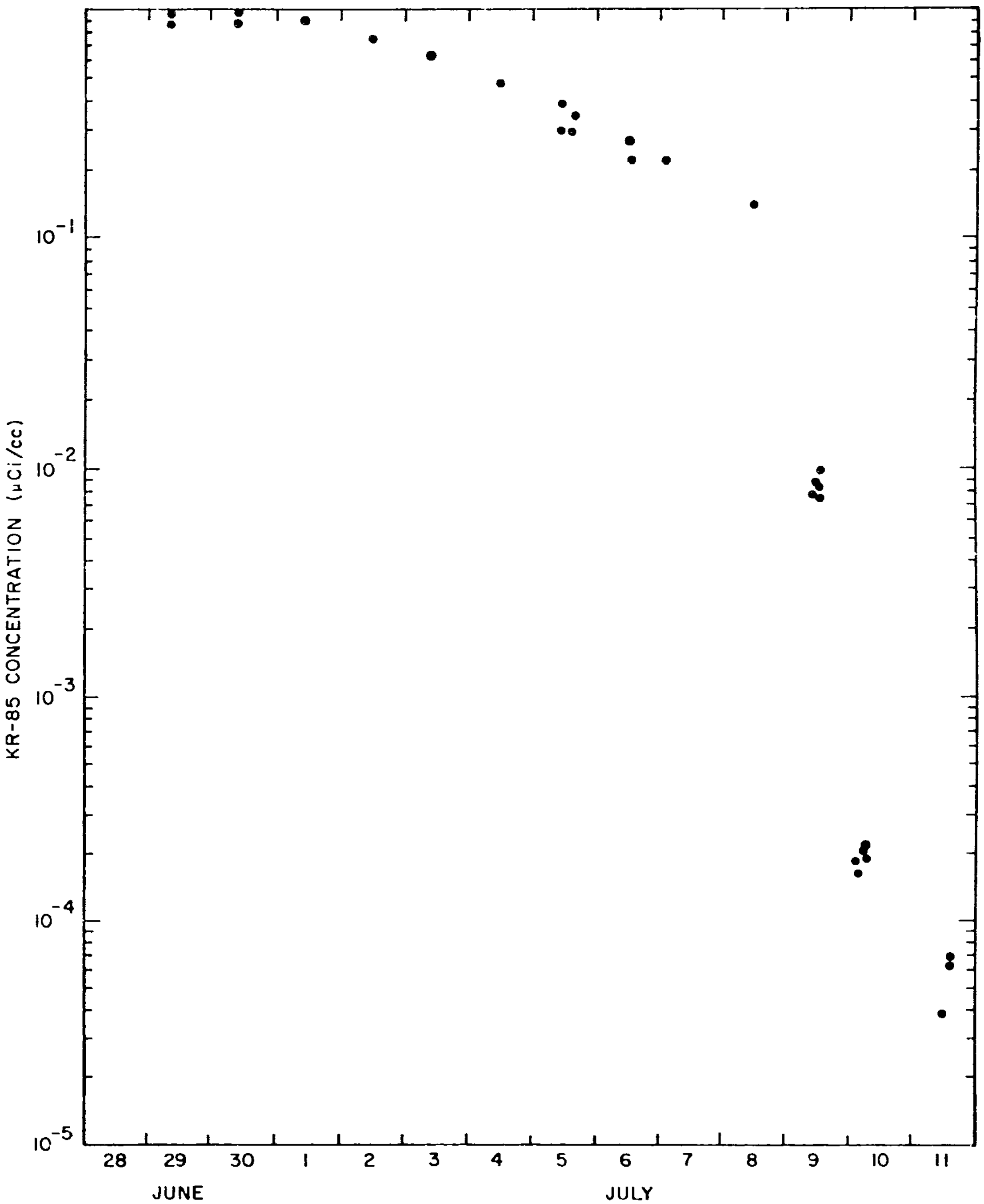

Figure 4 Reactor Building Atmosphere, Kr-85 Samples.

TABLE 21. IODINE 129 AND PARTICULATE CONCENTRATIONS FOR THE TMI-2 CONTAINMENT DURING VENTING - SAMPLES TAKEN FROM R-626 PENETRATION[26]

	Values in µCi/cc				
PERIOD	TOTAL I-129	Cs-134	Cs-137	Sr-90	Sr-89
7/4-7/6	2.76 E-11	5.87 E-12	4.05 E-11	7.0 E-12	-
7/6-7/8	1.46 E-11	2.84 E-12	1.80 E-11	6.4 E-12	-
7/8-7/9	4.80 E-12	1.74 E-11	1.21 E-10	2.6 E-11	<1.1E-12
7/10-7/12	1.82 E-12	1.95 E-11	1.40 E-10	4.67 E-11	<1.6 E-12

3.6.3.1 Station Vent Monitors. As indicated in Section 3.2.3, HPR-219A provided a printed record of particulate, iodine, and gaseous releases. During the venting period, an auxiliary operator (AO) was stationed full time at the HPR-219A terminal located on the turbine deck just outside the Control Room. The AO punched out instantaneous readings upon request (AO and Control Room communicated via two-way radios) and also printed out 10-minute and hourly averages. Additionally, the HPR-219A particulate filter and charcoal cartridge were changed daily and submitted for gross beta, Ge(Li), gross alpha, and Sr-89/90 analysis and Ge(Li) analysis, respectively. The particulate filters and charcoal cartridges were also sent to the EPA for confirmatory analyses. Replacement of the particulate filter and charcoal cartridge required venting shutdown. This was at first done at noon on each day but later the filter changes were scheduled at midnight.

The daily and cumulative curies of Kr-85 released during the venting period as computed directly from HPR-219A and station vent flow rate readings are listed in Table 22 (see Section 3.7 for the corrected total curies of Kr-85 released). An hourly record of Kr-85 release activity, station vent flow rate, the Kr-85 curie release rate, and the total number of curies of Kr-85 released is presented in Appendix A for the entire venting period. (Note: Appendix A has utilized correction factors to HPR-219A and stack flow rate different than those discussed in Section 3.7. This accounts for the small difference in estimated curies of Kr-85 released.)

The analyses of the particulate filters from HPR-219A showed that at no time did the particulate level approach the 6 E-10 μCi/cc gross beta-gamma limit that would have required shutdown of venting. Following the cessation of venting the particulate filters were sent to Teledyne Isotopes Inc. for more extensive analyses. After batching all the particulate filters, Teledyne performed specific chemical separation for Cs-137 and Sr-90. The results showed that during the period from 0754 hours on June 28, 1980 through 1200 hours on July 11, 1980, 5.50 E-6 curies of Cs-137 and 5.72 E-9 curies of Sr-90 were released. Gross analyses for alpha and beta-gamma were also performed on the composite of the particulate filters. These results showed 1.59 E-6 curies of gross alpha (not otherwise identified) and 1.24 E-6 curies of gross beta-gamma were released during the venting period[27]. The analyses of the charcoal filters from HPR-219A done onsite and confirmed by EPA showed no detectable levels of iodine.

TABLE 22. DAILY TOTALS FOR Kr-85 PURGE FROM JUNE 28, 1980 to JULY 11, 1980

DATE	DAILY CURIES	TOTAL CURIES
6/28	266	266
6/29	974	1,240
6/30	3,056	4,296
7/1	4,988	9,284
7/2	3,728	13,012
7/3	4,246	17,258
7/4	4,343	21,601
7/5	2,930	24,531
7/6	2,836	27,367
7/7	1,989	29,356
7/8	3,643	32,799
7/9	1,376	34,375
7/10	29	34,404
7/11	10	34,414

Reference 27 also reported that an estimated 1.3 curies of tritium had been released between 0754 hours on June 28, 1980 and 1200 hours on July 11, 1980. The tritium was monitored by continuous bubbling of the gaseous effluent through a water volume, aliquots of which were analyzed by a liquid scintillation counting technique.

As mentioned previously, the HPR-219A particulate channel readings were adversely affected by the background Kr-85 activity. Thus, two alternative particulate monitoring systems were installed and utilized. Both systems are described in Section 3.2.3. The HPR-219 bypass particulate filter system required the filter to be changed every 15 minutes. The filters were immediately submitted for gross beta, gross alpha, and Ge(Li) analysis. Confirmatory analyses were performed by the EPA. This system was utilized through the initial days of venting and again was used several times when the alternative real-time particulate monitoring system (HPR-219B) failed. The results of the analyses of the 15-minute filter samples were used to calculate particulate release rates. All of the sample results gave release rates less than 10% of the Technical Specification limit of 0.3 μCi/sec. The highest calculated release rate was 3.40 E-3 μCi/sec which occurred near midnight on July 3, 1980. This release rate is 1.13% of the Technical Specification limit. No specific isotope or group of isotopes were identified as being responsible for this increase. Following this peak the particulate activity returned to a normal level at about 0430 hours the same day. Most measurable release rates were a factor of 10 below this peak valve or about 0.12% of the Technical Specification limit.

The HPR-219B real-time particulate monitoring system had its equipment and output located beside the HPR-219A terminal on the turbine deck. Readings were taken by the AO monitoring HPR-219A at 1000-second intervals and the change from the last reading recorded. The readings remained essentially the same (between 95 and 155 cpm) with no increase between readings greater than 47 cpm. The action level was an increase of 150 counts over the previous 1000-second reading which is equivalent to a stack concentration of 5.8 E-10 μCi/cc or one-tenth of the instantaneous particulate release rate Technical Specification limit.

3.6.3.2 HPR-229. HPR-229 is the radiation monitor located on the exhaust of the MHCS. HPR-229 includes a particulate, iodine, and gas channel. In addition, for Reactor Building venting, a high range gas channel was added. Indication and a stripchart record of HPR-229 readings were provided from Control Room Panel No. 12. HPR-229 readings were also reported in logs kept by the CRO and Shift Engineer.

The particulate filter tape from HPR-229 was removed daily during venting concurrently with the changeout of the HPR-219A filters and submitted for Ge(Li) analysis. The filter tapes removed on June 29 and 30 were also submitted to the EPA for confirmation analysis. Although detectable levels of Cs-134 and more frequently Cs-137 were found, no level was high enough to cause any concern and all were well below Technical Specifications. The HPR-229 charcoal cartridge was removed once during venting on July 7, 1980 and showed no detectable levels of iodine.

3.6.3.3 HPR-226. HPR-226 is the radiation monitor located on the exhaust of the MRBPS. HPR-226 includes a particulate, iodine, and gas channel. Indication and a stripchart record of HPR-226 readings were provided from Control Room Panel No. 12. HPR-226 readings were also reported in logs kept by the CRO and Shift Engineer.

3.6.3.4 Other Effluent Radiation Monitoring. In addition to the effluent radiation monitoring described above, some other effluent radiation monitoring was conducted during venting. This monitoring included:

- Daily gas samples taken from HPR-219A for Kr-85 analysis. These samples were correlated to HPR-219A readings (see Section 3.7).
- Stack sampling to establish the radiation profile of the stack. This effort helped to find the reasons for the discrepancies between monitored releases and the estimated Kr-85 inventory (see Section 3.7).
- Gas samples of the MHCS filter plenum (through DPI 8088) and exhaust (through the HPR-229 sample line) to determine the cause of the difference in measured Kr-85 releases and the changing Reactor Building Kr-85 inventory (see Section 3.7).

3.7 Analysis

The major concern that arose as a result of the venting operation was the large difference between the measured curies of Kr-85 released and the original estimated inventory of Kr-85, i.e., 34,414 curies versus 57,000 curies. To account for this discrepancy, an investigation was conducted by Met-Ed/GPU into the potential errors (systematic or random) associated with the following parameters upon which the estimated Kr-85 in the Reactor Building and measured Kr-85 vented are based:

1. Reactor Building Kr-85 concentration
2. Reactor Building free volume
3. Plant stack gas velocity
4. Plant stack Kr-85 concentration

As summarized below, the Met-Ed/GPU investigation resolved the above discrepancy. The complete documented results of the Met-Ed/GPU investigation are reported in Reference 28.

Reactor Building Kr-85 Concentration — Reactor Building atmosphere samples prior to and during venting were taken as described in Section 3.2.3. Although the sampling method provided repeatable results, the accuracy of the sampling method had not been established using National Bureau of Standards (NBS) traceable Kr-85 gas standards. Prior to the purge a comparison was made between Reactor Building samples taken in the 30 cc sample bulb used by Met-Ed/GPU and an NRC 32.65 cc spherical sample container, and an NBS certified standard for Kr-85 in a 32.65 cc spherical sample container. The comparison showed a bias error of about +0.2 μCi/cc or that the original Reactor Building Kr-85 concentration was about 0.8 μCi/cc and not 1.04 μCi/cc. This was further supported by data taken during the purge when 1640 cc Marinilli samples were taken instead of the 30 cc sausage samples. The use of Marinelli beakers, which allowed Reactor Building samples to be compared to availabe 1640 cc NBS traceable Kr-85 gas standards, could not be used initially because concentrations higher than 0.35 μCi/cc are above the capability of the on-site counting equipment.

Reactor Building Free Volume -- The original estimates of the total number of curies of Kr-85 in the Reactor Building were based on an estimated Reactor Building volume of 2 E6 $ft.^3$ A more refined estimate of the Reactor Building free volume gave a value of 1.97 ± 0.02 E6 $ft.^3$ or essentially the same as the previous estimate. The Met-Ed/GPU investigation also concluded that the possibility of Kr-85 gas being trapped in pockets of the Reactor Building was unlikely.

Plant Stack Gas Velocity -- It was recognized early in the purge that stack flow, computed from a measurement of stack velocity, was not correct because the flow was low compared to the sum of flows entering the stack and varied depending on the time-of-day (low during the day -- high during the night). Therefore, during the purge, a stack velocity traverse was made as part of an effort to correct this error. Based on this stack velocity traverse and the difference between the measured stack flow and the sum of building flows exhausting into the stack, Reference 28 showed that there was a low plant stack flow measurement error of between 6.2% and 13.2%. Therefore, measured plant stack gas flow should be multiplied by a factor of 1.097 $\pm$ 0.035 to compensate for this error and obtain an accurate measurement of Kr-85 released.

Plant Stack Kr-85 Concentration -- The concentration of Kr-85 exiting the plant stack was measured by HPR-219A during the purge (see Section 3.2.3). To determine the accuracy of the Kr-85 concentration being reported by the HPR-219A radiation monitor, 1640 cc Marinelli beaker gas samples were taken in the stack and at the HPR-219A monitor. The samples were drawn in Marinelli beakers so they could be compared to a 1640 cc NBS certified gas standard. A comparison of the stack Marinelli sample result (2.98 E-4 μCi/cc) with the HPR-219A reading at the time (2.31 E-4 μCi/cc) showed a 29% difference. This difference was attributed to the pressure difference (-3.0 psi) between the stack (14.7 psia) and the sample line at the HPR-219A gas monitor (11.7 psia). This pressure difference is primarily due to the HPR-219A particulate filter. Since this filter was changed daily, the pressure difference varied daily from -0.5 psi for a new filter to a maximum of -3.0 psi after one day's use. Assuming a linear relation between the pressure difference and the HPR-219A monitor reading, the stack concentration should be multiplied by a factor of 1.167 $\pm$ 0.121 to obtain a more accurate estimate of the number of curies which were actually released from the plant stack.

Conclusions -- Based on the above findings, the initial amount of Kr-85 contained in the Reactor Building (Reactor Building Kr-85 concentration times Reactor Building free volume) is estimated to range from 43,000 to 46,200 curies with a median value of 44,600 curies. The measured amount of Kr-85 vented (plant stack gas flow times plant stack Kr-85 concentration) corrected for the stack flow and stack Kr-85 concentration errors (1.097 $\pm$ 0.035 and 1.169 $\pm$ 0.121, respectively) is estimated to range from 38,302 to 50,254 curies with a median value of 44,132 curies. The downward revision of the initial Reactor Building Kr-85 inventory and upward revision of vented Kr-85 has led to elimination of the discrepancy since the initial Reactor Building inventory is now enveloped by the measured Kr-85 vented.

4.0 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAMS

The several radiological environmental monitoring programs on-going in conjunction with purging the Reactor Building atmosphere, constituted perhaps the most extensive monitoring effort ever instituted at a commercial nuclear facility. The monitoring programs established following the March 28, 1979 accident were supplemented and expanded to insure effective monitoring of Kr-85 levels in the environment and to serve as a real-time method of verifying off-site dose predictions. Additionally, one of the principal concerns was in establishing a monitoring program which had credibility with the general public and which could accurately measure and expeditiously report its finding. The principal organizations involved in the monitoring programs were the U. S. Environmental Protection Agency (EPA), the U. S. Nuclear Regulatory Commission (NRC), Metropolitan Edison/General Public Utilities (Met-Ed/GPU), the U. S. Department of Energy (DOE), the Pennsylvania Department of Environmental Resources (DER), Pennsylvania State University, and the U. S. Public Health Service. There also existed a Citizens Radiation Monitoring Program.

To rapidly disseminate to the public results of the environmental monitoring programs, daily news conferences were held by EPA and daily news releases were published by EPA and Met-Ed/GPU. Additionally, DER published daily news releases of the results of the Citizens Radiation Monitoring Program and attended the daily EPA news conferences along with the NRC to report these findings.

The following sections provide a description of the pertinent radiological environmental monitoring programs[a] that were established or existed during the venting of the Reactor Building and their measured or computed findings. In cases

a. Because (1) Kr-85 was the dominant radionuclide contained in the Reactor Building with all other radionuclides below minimum detectable limits (e.g., radioactive isotopes of Xe and I and other Kr isotopes) or below maximum permissible concentrations (e.g., Cs-137, Sr-89/90), because (2) all releases were through HEPA filters (99.9% efficient or better) to reduce any release of particulate radiation to negligible quantities, and because (3) Kr-85, being a noble gas, has no significant food pathway involvement, the only important monitoring activities were those capable of detecting Kr-85. Hence, not all of the environmental radiation monitoring activities which were being conducted during the venting are discussed in this section. Particulate and radioiodine monitoring are addressed to some extent, however, since they were used to verify that these releases were negligible.

where the responsible organization has published a report containing the accumulated monitoring data and results, this report will be referenced and only the significant findings and conclusions will be reported here.

4.1 U.S. Environmental Protection Agency (EPA)

EPA has been designated by the Executive Office of the President as the lead federal agency for conducting a comprehensive long-term environmental radiation monitoring program as a follow-up to the March 28, 1979 accident at TMI-2. The long-term environmental radiation surveillance plan for TMI jointly developed by EPA, NRC, the Department of Health and Human Services (formerly HEW), DOE, and the Commonwealth of Pennsylvania is presented in Reference 29. In addition to this long-term monitoring program, EPA with the assistance of DOE, the U. S. Public Health Service, and the Nuclear Engineering Department of Pennsylvania State University conducted an expanded monitoring program just for the venting of Kr-85 from the Reactor Building. A description and the results of the radiological environmental monitoring program important to monitoring the Reactor Building venting for which EPA was responsible or coordinated is presented below. For additional information concerning the results of the EPA monitoring effort and a discussion of these results see Reference 67.

To gain an appreciation of the additional amount of effort expended for the environmental radiation monitoring efforts described here, consider that during the venting period, EPA brought in 24 persons to augment its permanent six-member staff. Other federal personnel added included three officials of the U. S. Public Health Service, a four-member helicopter crew from DOE, and a crew to launch weather balloons at Harrisburg International airport. Also, gathering weather data involved a field crew in the TMI area and running ARAC (see below) required a nine-person computer support staff at Lawrence Livermore Laboratory in California. The EPA Middletown office staff was also supplemented with 13 senior citizens who carried out various clerical duties. And finally, the noble gas monitoring activities conducted by Penn State involved approximately 10 persons.

4.1.1 Surveillance Stations

Description

EPA as part of its long-term surveillance program operates a network of eighteen continuous air monitoring stations shown on Figure 5 and listed in Table 23. These stations operate at radial distances ranging from one-half mile to seven miles from TMI. Seven miles was established as the point well beyond that which EPA expected to detect any emissions from TMI-2. Due to the proximity of Hill Island to Three Mile Island, a continuous air monitoring station was additionally placed there for operation immediately prior to and during Kr-85 venting. Each of these 19 stations includes an air sampler, a gamma rate recorder, and normally three, but for venting four, thermoluminescent dosimeters (TLDs).

The air sampler units pull air at approximately two cfm through a glass-fiber filter for particulate (i.e., Co-58 and -60, Cs-134 and -137, and Ru-106) detection and then through activated charcoal filters for radioiodine detection. For the venting period the glass-fiber filters and activated charcoal filters were collected daily and evaluated immediately. Analysis was by high resolution gamma spectrometry at EPA's Middletown laboratory using a Ge(Li) detector. Sensitivity for a 10-minute counting period on the Ge(Li) detector is 3 E-13 μCi/cc for an average sample. The particular filters were saved and a one-half portion of each filter collected was composited for each sampling station and analyzed for Sr-89 and -90; U-235 and -238; Pu-238, -239, and -240. All charcoal cartridges collected during venting were composited for each sampling location and analyzed for I-129. Data from this air surveillance network was intended to document any low level releases of radionuclides other than Kr-85, should they have occurred.

Each of the 19 monitoring stations also contained a gamma rate recorder for measuring and recording external exposure and four TLDs. The gamma rate recorder charts were collected daily during Kr-85 venting along with the particulate filters and charcoal cartridges. Three TLDs normally located at the continuous air monitoring stations were exchanged just prior to venting during the normal quarterly exchange and were to be picked up at the next normal quarterly exchange at the end of September. In addition, one more TLD was added to each station at the normal

FIGURE 5. **Long Term Air Monitoring (Three Mile Island)**

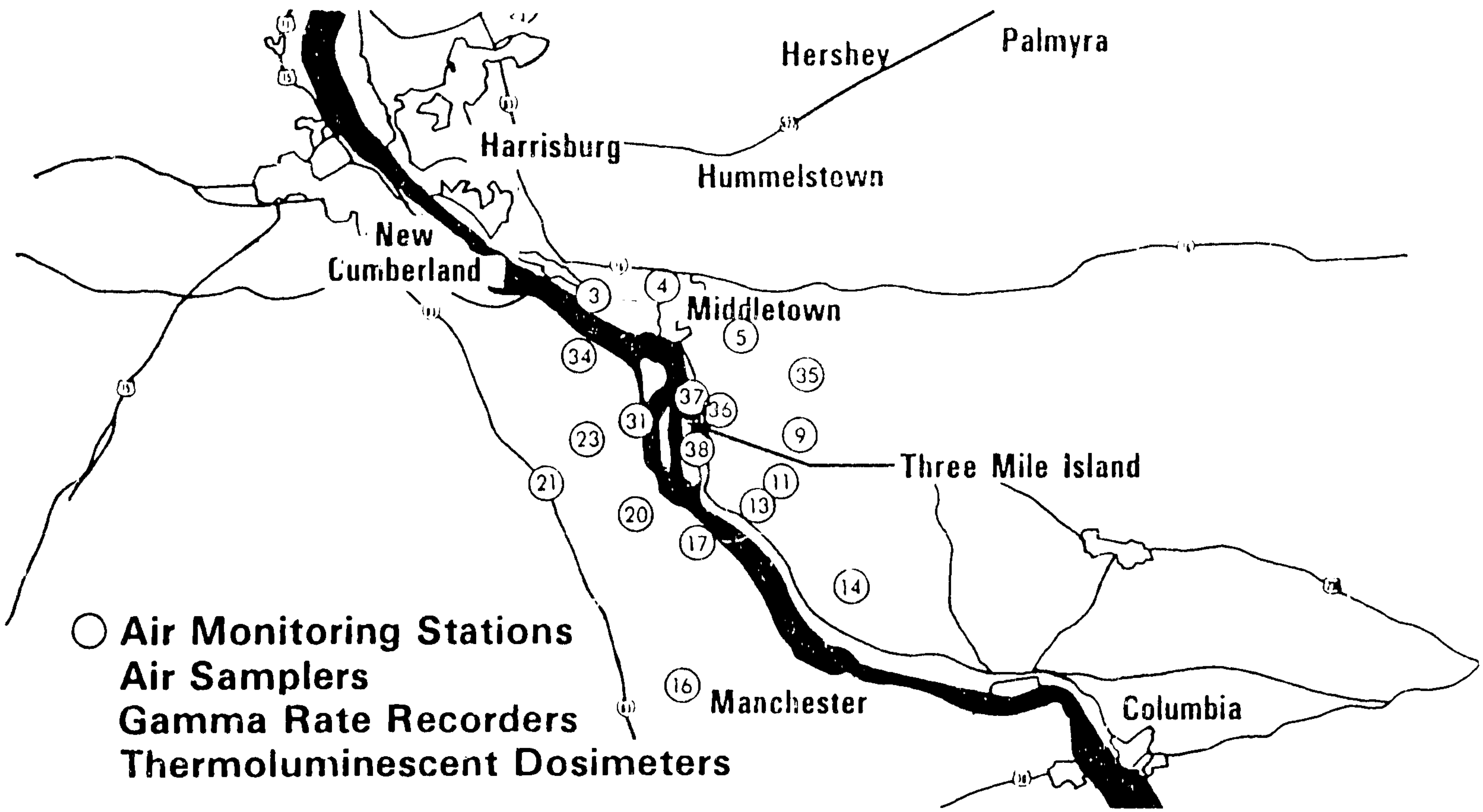

TABLE 23. THREE MILE ISLAND EPA LONG-TERM SURVEILLANCE STATIONS

(Air Samplers, Gamma Rate Recorders, TLD's)

STATION	AZ	DIST	ASSOCIATED TOWN
3	325	3.5	Meade Heights, PA -- Harrisburg Intl. Airport
4	360	3.0	*Middletown, PA -- Elwoods' Sunoco Station
5	040	2.6	Royaltown, PA -- Londonderry Township Building
9	100	3.0	Newville, PA -- Brooks Farm
11	130	2.9	Falmouth, PA -- Charles Brooks Residence
13	150	3.0	Falmouth, PA -- Dick Libhart Residence
14	145	5.3	*Bainbridge, PA -- Bainbridge Fire Company
16	180	7.0	*Manchester, PA -- Manchester Fire Department
17	180	3.0	*York Haven, PA -- York Haven fire Station
20	205	2.5	Pleasant Grove, PA -- Zane Reeser Residence
21	250	4.0	*Newberrytown, PA -- Exxon Kwick Service Station
23	265	2.9	Goldsboro, PA -- Muellar Residence
31	270	1.5	*Goldsboro, PA -- Dusty Miller Residence
34	305	2.7	Plainfield, PA -- Polites Residence
35	068	3.5	Royaltown, PA -- George Hershberger Residence
36	095	0.5	TMI Observation Center
37	025	0.7	North Gate, TMI
38	175	0.8	South Gate, TMI

*Sampling stations located in indicated town. Other sampling stations are located near indicated towns.

quarterly exchange just prior to venting and was removed immediately after venting was complete. The TLDs are read at EPA's Environmental Monitoring Systems Laboratory in Las Vegas, Nevada. Neither the gamma rate recorders or the TLDs, which were designed to measure gamma radiation exposure only, were expected to record any effect from the venting of Kr-85 from the Reactor Building at the levels expected to occur.

EPA also had five fixed noble gas sampling stations (Bainbridge, Goldsboro, Hill Island, Middletown, and the TMI Observation Center) operational immediately prior to and during the venting. The locations were chosen to provide representative coverage with emphasis on the predominant wind directions. The noble gas samples which consisted of pressurized tanks of compressed air each of which contained at least 0.6 standard cubic meters of air were picked up daily and analyzed immediately in laboratory space provided by DER in Harrisburg. Sample analysis consists of a combination of cryogenic and gas chromatographic techniques for the quantitative separation of gases after the water vapor and carbon dioxide are removed. The krypton and xenon are adsorbed on activated charcoal and are then removed one at a time into evacuated liquid scintillation vials. Degassed liquid scintillation cocktail is added to the vials. The radioactivity in the vials is then determined using a liquid scintillation counter. The detection limit for this method is about 4 E-12 μCi/cc for each gas. Although the turn-around time precludes the use of these samples as "real-time" monitors, they did provide documentation of extremely low concentrations of Kr-85.

In addition to the TLDs at the 19 monitoring stations, EPA had similar TLDs (gamma sensitive only) at 0.25 mile intervals along roads immediately parallel to the Susquehanna River near TMI out to a distance of about 2.5 miles from TMI-2. TLDs were also located on Shelley, Hill, Henry, Kohr, and Beech Islands located 0.5 to 1.5 miles west of TMI-2. These dosimeters are read quarterly.

<u>Results</u>

As reported by EPA in Reference 67, Kr-85 levels during the 14 day venting period provided a maximum total skin dose of 0.86 mrem at the TMI Observation Center based on the noble gas samples taken there. The TMI Observation Center is at an azimuth of 95^{o} and is approximately 0.5 mile from the release point. This 0.86 mrem is about six percent of the skin dose limit of 15 mrem. At the other stations, the total skin dose accumulated since venting began was 0.014 mrem at Bainbridge, 0.019 mrem at Goldsboro, 0.049 mrem at Hill Island, and 0.079 mrem at Middletown. The accumulated whole body doses at the five noble gas sampling stations were all very small fractions of the 5 mrem per year whole body dose standard with the peak dose of 0.0071 mrem again occurring at the TMI Observation Center.

The gamma spectral analysis performed on the daily air samples (air filter plus charcoal) detected no activity above detectable limits. The preliminary results from the radiochemical analysis for Sr-89 and -90, U-235 and -238, and Pu-238, -239, and -240 also suggest that none of these radionuclides were present in levels greater than those measured in the area prior to venting. These samples are now being re-analyzed to resolve several anomalous results. I-129 results from the activation analysis of the composited charcoal cartridges are not yet available.

Evaluations of the charts from the gamma rate recorders at each of the air monitoring stations and of the TLD's deployed during the purge period showed gamma exposure levels within the normal background range.

4.1.2 Personnel Dosimeters

TLDs voluntarily worn by some 50 residents of the off-site area surrounding TMI were exchanged immediately before and after the Reactor Building venting. Based on these TLDs a measure of the total gamma dose to which these individuals were exposed during the entire period of venting were obtained. The results showed that gamma exposure levels to these persons were within the normal background range.

4.1.3 Mobile Monitoring and Sampling - EPA

Description

During the entire period of venting, EPA had two mobile monitoring teams operating from vehicles that had 2-way radio communication with the EPA office in Middletown. These teams were positioned at locations where the highest concentrations of radionuclides were expected to occur, as predicted by the U. S. Department of Energy's Atmospheric Release Advisory Capability (ARAC).

ARAC is a real-time, computerized atmospheric dispersion model that used wind speed and direction data obtained from weather balloons released from Harrisburg International Airport and all available terrain and surface meteorologic data within 30 kilometers (18.5 miles) of TMI. The model was exercised hourly by personnel from Lawrence Livermore Laboratory in California. During the first several days of venting, the U.S. Department of Energy's Aerial Measuring System, flown in a helicopter out of Capital City Airport, obtained independent measurements of Kr-85 and other radionuclides, if they could be detected, to confirm the accuracy of the ARAC predictions.

In order to provide two team members to man each of EPA's two mobile monitoring units 24 hours a day during the venting period, U.S. Public Health Service personnel supplemented those from EPA.

Each EPA vehicle was equipped with two portable radiation survey instruments. These were:

(1) An Eberline PAC4G which is a constant flow proportional survey meter. This instrument, fitted with the appropriate detector, will detect 0.1 MPC (Maximum Permissible Concentration) which is equal to 3 E-8 μCi/cc of air.

(2) A Ludlum Model 2a with an HP2A pancake probe which will detect 50% of MPC or 1.5 E-7 μCi/cc of air.

The portable survey instruments were used to verify the team's position within the plume. Monitoring personnel recorded the survey meter readings on a log and turned in their logs to the EPA operations center at the end of their 12-hour shifts. Generally, they took readings at 15-minute intervals if only background was being measured and at 5-minute intervals or more frequently if they obtained readings above background. Results above background were radioed into the EPA office in Middletown.

Each EPA vehicle also carried an electrical generator to power a noble gas sampler, an atmospheric sampler for measuring tritium as HTO, and an air sampler identical to the one used at the 19 fixed locations previously discussed. The noble gas sampling equipment collected samples of filtered, compressed air in pressurized tanks with each sample containing at least 0.6 standard cubic meters of air. Sampling for tritium was accomplished by passing filtered air through a molecular sieve which absorbs all the moisture in a given volume of air. Typically five cubic meters of air are passed through a molecular sieve over a seven day sampling period. The HTO or tritiated water absorbed on the molecular sieve is recovered from the column by distillation.

To assure that its noble gas sampling was representative of maximum krypton exposures in each of the 16 (22 1/2^{o}) sectors, EPA assigned compressed air tanks carried by the mobile teams to each sector. The monitors collected air in a given tank only when the plume from TMI was predicted by the ARAC to be most concentrated in the location where they were sampling. Monitors kept a log of the time each tank was used. As a result, the number of samples for each sector varied within the range of zero to three total samples.

Whenever the EPA monitors were operating the noble gas sampler, they operated the atmospheric moisture sampler and the air sampler no matter which sector the mobile team was in. The atmospheric moisture sample from each mobile unit was collected weekly for analysis, whereas the mobile unit's air sampler particulate filter and charcoal cartridge were collected and analyzed only at the end of the venting operation.

The particulate filters and charcoal cartridges from the two EPA mobile units were analyzed by high resolution gamma spectometry at EPA's Middletown laboratory just like the particulate filters and charcoal cartridges from the 19 fixed monitoring stations (see Section 4.1.1). After the filters were analyzed, half portions of each were analyzed for Sr-89 and -90, U-235 and -238, and Pu-238, -239 and -240. The charcoal cartridge from each of the two mobile units was also analyzed for I-129 after the gamma spectral analysis was completed.

Following the completion of venting, the compressed air sample bottles collected by the mobile units were analyzed exactly as the fixed noble gas samples had been (see Section 4.1.1). In cases where the sample bottles had not been completely filled, they were filled to the prerequisite pressure required to enable analyses. An empty bottle was also filled at the same time to serve as a background sample.

The atmospheric moisture samples collected over week-long periods by the EPA mobile monitoring teams were analyzed for tritium by liquid scintillation techniques. The sensitivity of the procedure is 4 E-7 μCi/cc of water.

Results

As reported in the daily issues of EPA's "Environmental News" (References 30 - 42) the two EPA mobile teams were often able to measure radiation levels above background with their portable radiation survey equipment when located in the plume. The results of the mobile noble gas sampling performed in each of the 22 $1/2^{\circ}$ sectors are shown in Table 23A. The values for the E and ESE sectors are listed as a range because one sample was inadvertently collected across the sector boundary. The value for that sample is assigned totally to each sector to establish the upper limit. The lower limit is defined by deleting that sample from each sector.

The analysis results of the air samples (air filter and charcoal) taken by the mobile teams were reported with the results from the 19 fixed stations in Section 4.1.1. The results of EPA's tritium analysis of the atmospheric moisture samples, one collected by each of its mobile monitoring units between June 28 and July 6, 1980 and between July 6 and July 11, 1980, ranged from 1.9 to 8.4 E-12 μCi/cc air and from 210 to 400 in Tritium Units. The tritium concentrations in μCi/cc air were somewhat dubious, however, because the amount of water vapor recovered from the molecular sieve collectors was inconsistent, and lower than would have been expected from the relative humidity. A comparison of the results expressed in Tritium Units with normally expected tritium levels indicate tritium concentrations somewhat above the expected ambient background, but the radiation dose equivalent to the critical organ, the total body fluids, was insignificant, less than 0.001 mrem.

4.1.4 Mobile Monitoring and Sampling - Penn State

Description

The Nuclear Engineering Department of Pennsylvania State University (Penn State) also had a mobile monitoring team in the off-site area during krypton venting. The Penn State vehicle was equipped with a compressed air sampler for collecting "grab" samples over 18-minute periods. These samples were analyzed by Penn State for Kr-85 concentration using the Penn State noble gas monitor.

TABLE 23A. MOBILE NOBLE GAS SAMPLING RESULTS[67]

Sector	Skin Dose (mrem)	Whole Body Dose (mrem)
N	0.39	0.0032
NNE	0.18	0.0015
NE	0.33	0.0028
ENE	0.27	0.0023
E	0.15-0.35	0.0013-0.0029
ESE	0.16-0.36	0.0013-0.0030
SE	0.042	<0.001
SSE	0.03	<0.001
S	No Sample	
SSW	0.001	<0.001
SW	No Sample	
WSW	0.016	<0.001
W	<0.00001	<0.00001
WNW	Sample Lost In Analysis	
NW	<0.001	<0.001
NNW	0.13	0.0011

In the Penn State program sample air was compressed into gas cylinders to 3,000 psig using a portable scuba compressor mounted in a van. The bottles of sampled air were then transported to EPA's laboratory in Middletown where the Penn State noble gas monitor system was located. The sample bottle was connected to the 12 inch diameter steel sphere of the monitor system. Upon opening the interconnecting valves, the pressure was allowed to equalize between the two pressure chambers, thereby pressurizing the spherical sample chamber to about 1150 psig. A high resolution Ge(Li) detector located at the center of the sphere was then used to detect the radionuclides in the gas with data accumulation and subsequent processing performed by a multichannel pulse height analyzer. The Penn State system has a detection limit of approximately 3 E-8 μCi/cc of Kr-85 and allowed quick (one to three hour) turnaround times for identification and resolution of airborne radiation.

The Penn State team was positioned by EPA via two-way radio at locations near and downwind from the EPA mobile teams to provide an independent check of the measurements EPA's teams obtained. Also, the Penn State team was periodically requested to collect samples at populated locations away from the immediate vicinity of TMI to help assure the public in these locations that they were not being exposed to any significant radioactivity.

A final report completely describing the Penn State Kr-85 monitoring program during the purge is contained in Reference 66.

Results

The results of the Penn State sampling program were summarized daily in the EPA publication "Environmental News" (References 30 - 42). Of the 124 samples taken and analyzed during the purge period, 37 were determined to contain Kr-85 above the lower limits of detection. The measured concentrations ranged from 1.5 E-6 to 3 E-8 μCi/cc. All samples collected in the communities of Elizabethtown, Marietta, Newberrytown, East Manchester, Fairview, Mount Joy, Lancaster, Columbia, York, and York Haven contained no detectable concentration of Kr-85. For the complete results of the Penn State Kr-85 monitoring efforts, see Reference 66.

4.2 Citizens Radiation Monitoring Program

The creation of the Citizens Radiation Monitoring Program evolved independently of the Reactor Building Kr-85 venting plans, but its operation during the venting period added an extra element of credibility to the overall radiological environmental monitoring efforts. The Citizens Radiation Monitoring Program was spawned by the expressed interest of the citizens and local governmental entities and was supported by the efforts of the Pennsylvania Department of Environmental Resources (DER), DOE, EPA, and Pennsylvania State University (Penn State). The DER acted as the coordinator and principal interface with the citizen monitors and local government officials. Penn State provided the training program for the selected citizen monitors. DOE provided funds for the training program and DOE and EPA provided the radiation monitoring equipment utilized and technical support personnel.

After the consensus had been reached to form a Citizens Radiation Monitoring Program, DER spoke to first the county commissioners of Lancaster, York, and Dauphin counties and then to the local township and community officials. From the townships and communities within a five mile circle of TMI, 12 monitoring stations were established (see Table 24). Citizens selected by their local officials to monitor these stations then attended a series of one Sunday and ten evening training sessions conducted by Penn State. The topical outline for the training program which included instruction in the basics of radiation, its effects, detection techniques, and also hands on experience with monitoring equipment in the field is provided in Table 25. A total of 49 persons including alternates graduated from the training program by passing an examination demonstrating their competence in both the theoretical and practical aspects of the course.

Regular daily monitoring at the 12 monitoring stations began approximately one month prior to the commencement of venting. Each monitoring station was equipped with a Lear Siegler (LSI) gamma rate recorder used to measure gamma radiation levels at the monitoring site. These recorders are sensitive enough to measure radiation from naturally occurring radiation sources. Each monitoring station was also equipped with a Ludlum (pancake) beta rate recorder used to measure beta and gamma radiation levels at the monitoring site. These recorders are also sensitive enough to measure radiation from naturally occurring radiation sources.

TABLE 24. CITIZENS RADIATION MONITORING STATION LOCATIONS

Municipality	Azimuth	D (mi)
Londonderry	40°	1
Elizabethtown	90°	6.5
West Donegal	100°	7
Conoy	160°	2
East Manchester	170°	7
York Haven	175°	3
Newberry	245°	4.5
Goldsboro	270°	1.5
Fairview	285°	7
Lower Swatara	335°	2.5
Middletown	350°	2
Royalton	355°	5

TABLE 25. TOPICAL OUTLINE FOR TRAINING PARTICIPANTS FOR THE CITIZENS RADIATION MONITORING PROGRAM

DATE			TIME
April 2	A.	Introduction to the Citizens Radiation Monitoring Program	3 hours
	B.	Radioactivity 1. Introduction and Definition of Terms 2. Radioactive Decay 3. Conservation Laws 4. Background Radiation and Sources	1 hour
April 7	C.	Interaction of Radiation with Matter 1. Introduction and Definition of Terms 2. Interaction Mechanisms	1.5 hours
	D.	Methods of Radiation Detection 1. Introduction and Definition of Terms 2. Detector Types 3. Detector Sensitivities	1.5 hours
April 8	E.	Radiation Counting Variables 1. Introduction and Definition of Terms 2. Systematic and Statistical Variables	1.5 hours
	F.	Laboratory Experiment GM Counting Experiment	1.5 hours
April 9	G.	Radiation Protection Units 1. Activity 2. Exposure Dose 3. Absorbed Dose 4. Equivalent Dose	1.5 hours

TABLE 25 (cont'd)

DATE			TIME
April 9-10	H.	Laboratory Experiment 1. Monitoring Equipment 2. Familiarization of Argon-41 Monitoring	6 hours
April 10	I.	Radiation Interaction in Biological Systems 1. Introduction and Definition of Terms 2. Radiation Effects 3. Regulations	1.5 hours
April 13	J.	Equipment Familiarization and Argon-41 Monitoring	5 hours
	K.	Laboratory Experiment Counting Statistics Laboratory	1.5 hours
April 14	L.	Citizens Radiation Monitoring Program 1. Purpose 2. Organization 3. Equipment 4. Procedures	1.5 hours
	M.	Three Mile Island Unit-2 1. The Accident 2. Proposed Methods of Cleanup	1.5 hours
April 15	N.	Supervised Area Monitoring	3 hours
April 16	O.	Supervised Area Monitoring	3 hours

TABLE 25 (cont'd)

DATE			TIME
April 18	P.	Final Exam	1.5 hours
	Q.	Discussion of Community Radiation Monitoring Results and Observations	1.5 hours
April 22	R.	Meteorological Considerations 1. Introduction and Definition of Terms 2. Atmospheric Conditions Affecting Dispersion	1.5 hours
	S.	Assignment of Personnel to Local Monitoring Teams	1.5 hours

Approximately one week prior to the start of Reactor Building venting, meetings were held with the citizen monitors to explain what they might expect to see. Since the beta rate recorder would be more sensitive to Kr-85, a concern level of 75 cpm above normal was established. If the level reached 125 cpm above normal the citizen monitor was directed to notify his or her local elected officials and the Technical Working Group (TWG). The TWG, consisting of representatives of DER, DOE/EG&G, EPA, and Penn State, would then contact EPA, Met-Ed/GPU, and NRC and also conduct any necessary confirmatory or follow-up actions.

The results of the Citizens Radiation Monitoring Program during the venting and for several days before and after were documented in daily press releases issued by DER[43-55] who was responsible for collecting and compiling daily the citizens monitoring reports (see Table 26) and stripchart recorder tapes from each of the 12 stations. These results indicated that at no time during the venting were gamma radiation levels above the normal background levels previously established. Table 27 summarizes the calculated beta skin doses from activity above the normal background of 0.005 mrem/hr as measured with the Ludlum beta rate recorders. As can be seen, the detected levels of radiation exposure were all small percentages of the 15 mrem per year skin dose limit. The largest accumulated dose was 0.105 mrem at the Londonderry station. All reported readings above normal background were consistent with the monitored wind direction and with readings taken by EPA and other agencies during the same time period. At no time was the concern level of 75 cpm above background reached at any station.

The Citizens Radiation Monitoring Program provided additional credibility and was therefore a positive addition to the overall radiological monitoring program during venting.

4.3 U. S. Nuclear Regulatory Commission (NRC)

The NRC did no special radiological environmental monitoring during the Reactor Building venting period. NRC does, however, operate one air sampling station located in the middle of the reactor complex where the particulate and charcoal filters are changed weekly and analyzed by gamma spectrometry. Two sets of TLD's at 59 locations are also maintained by NRC and both sets are read monthly. Each set contains two lithium borate and two calcium sulfate phosphers. The lithium borate phospher has the ability to detect beta radiation from Kr-85.

FORM-DER-[illegible]-490-TMI-4

TABLE 26

CITIZEN RADIATION MONITORING PROGRAM

MONITORING REPORT

ROYALTON

Date_____ _____

LSI (LEAR SIEGLER)		EBERLINE/LUDLUM (PANCAKE)	
Time on:		Time on:	
Time of reading:		Time of reading:	
Daily high:	mr/hr	Daily high:	~~mr/hr~~ cpm
Duration:	Minutes	Duration:	Minutes
Daily low:	mr/hr	Daily low:	~~mr/hr~~ cpm
Duration:	Minutes	Duration:	Minutes
Daily average:	mr/hr	Daily average:	~~mr/hr~~ cpm

Comments:_______________

Signature_______________
Citizen Recording Readings

Checked by:_______________

TABLE 27. CITIZENS RADIATION MONITORING PROGRAM
BETA RATE RECORDER RESULTS (MREM BETA SKIN DOSE)*

Location	June 26	June 27	June 28	June 29	June 30
Fairview	NB	NB	NB	NB	NB
Newberrytown	NB	NB	NB	NB	NB
Goldsboro	NB	NB	NB	NB	NB
York Haven	NB	NB	NB	NB	NB
East Manchester	NB	NB	NB	NB	NB
Lower Swatara	NB	NB	NB	NB	NB
Middletown	NB	NB	NB	NB	NB
Royalton	NB	NB	NB	NB	0.017
Londonderry	NB	NB	NB	NB	NB
Conoy	NB	NB	NB	NB	NB
West Donegal	NB	NB	NB	NB	NB
Elizabethtown	NB	NB	NB	NB	NB

NB = normal background (beta rate recorder readings less than 0.005 mrem/hour)

* The mrem beta skin dose shown in the table is the incremental beta skin dose above background.

TABLE 27 (cont'd)

Location	July 1	July 2	July 3	July 4	July 5
Fairview	NB	NB	NB	NB	NB
Newberrytown	NB	NB	0.003	NB	NB
Goldsboro	NB	NB	NB	0.004	NB
York Haven	NB	NB	0.037	NB	NB
East Manchester	NB	NB	NB	NB	NB
Lower Swatara	NB	NB	NB	NB	0.006
Middletown	NB	0.014	NB	NB	0.011
Royalton	NB*	0.019	NB	0.025	0.022
Londonderry	NB	0.024	0.056	0.015	0.004
Conoy	NB	0.004	NB	0.007	NB
West Donegal	NB	NB	NB	0.011	NB
Elizabethtown	NB	NB	NB	NB	NB

* A slight trace of Kr-85 was reported for a 10-minute period

TABLE 27 (cont'd)

Location	July 6	July 7	July 8	July 9	July 10
Fairview	NB	NB	NB	NB	NB
Newberrytown	NB	NB	NB	NB	NB
Goldsboro	NB	NB	NB	NB	NB
York Haven	0.004	NB	NB	NB	NB
East Manchester	NB	NB	NB	NB	NB
Lower Swatara	NB	NB	NB	NB	NB
Middletown	NB	NB	0.005	NB	NB
Royalton	NB	NB	0.007	NB	NB
Londonderry	0.006	NB	NB	NB	NB
Conoy	0.015	0.007	NB	0.003	NB
West Donegal	NB	NB	NB	NB	NB
Elizabethtown	NB	NB	NB	0.015	NB

TABLE 27 (cont'd)

Location	July 11	July 12	July 13	Total Accumulated
Fairview	NB	NB	NB	-0-
Newberrytown	NB	NB	NB	0.003
Goldsboro	NB	NB	NB	0.004
York Haven	NB	NB	NB	0.041
East Manchester	NB	NB	NB	-0-
Lower Swatara	NB	NB	NB	0.006
Middletown	NB	NB	NB	0.029
Royalton	NB	NB	NB	0.090
Londonderry	NB	NB	NB	0.105
Conoy	NB	NB	NB	0.036
West Donegal	NB	NB	NB	0.011
Elizabethtown	NB	NB	NB	0.015

The results obtained from the continuous air sampler for the period of May 20 to July 23, 1980 which includes the venting period are presented in Table 28. All I-131 and Cs-137 levels were below minimum detectable limits (approximately 5 E-14 µCi/cc) and no reactor related radioactivity was detected. Results of the environmental TLD measurements for the periods May 29 to July 2 (59 TLDs) and July 2 to July 31, 1980 (57 TLDs) found no gamma levels above background. Also, no detectable Kr-85 (beta radiation) was reported at a 95% confidence level (minimum detection limit approximately 150 MPC hours for Kr-85 beta).

4.4 Metropolitan Edison/General Public Utilities

All of the previously described radiological environmental monitoring efforts were not a substitution for, but an addition to the environmental surveillance plan of Met-Ed/GPU. The Met-Ed/GPU monitoring activities were a combination of the TMI-1 and -2 environmental technical specification requirements and the increased monitoring initiated after the March 28, 1979 accident. During the venting of the TMI-2 Reactor Building, Met-Ed/GPU also implemented a special radiation environmental monitoring program (REMP) to effectively monitor the off-site environment for releases of radioactive material particularly Kr-85. Special on-site and Auxiliary Building radiation monitoring programs were also instituted by Met-Ed/GPU. All three elements of Met-Ed/GPU's radiation monitoring efforts developed for the Kr-85 venting are discussed in the following sections. The effluent radiation monitoring program is addressed in Section 3.0.

4.4.1 Radiation Environmental Monitoring Program (REMP)

Concurrent with the licensing activities described in Section 2.0, Met-Ed/GPU developed a program to effectively monitor the off-site environment during the Reactor Building atmosphere purge. Analysis of the Reactor Building gas samples showed the Reactor Building atmosphere contained mostly radioactive Kr-85 with minute traces of particulates. Since the method of venting required that the effluent pass through high efficiency filters, the release of particulates would be negligible. Hence the primary emphasis in developing an environmental radiation monitoring program was to monitor the release of Kr-85 and this required environmental sampling techniques that were not then employed around TMI.

TABLE 28. NRC AIR SAMPLE RESULTS[56-63]

Sample	Period	I-131 (μCi/cc)	Cs-137 (μCi/cc)
HP-218	May 20 - June 4, 1980	< 5.1E-14	< 5.1E-14
HP-219	June 4 - June 11, 1980	< 5.2E-14	< 5.2E-14
HP-220	June 11 - June 18, 1980	< 5.4E-14	< 5.4E-14
HP-221	June 18 - June 25, 1980	< 5.2E-14	< 5.2E-14
HP-222	June 25 - July 2, 1980	< 4.9E-14	< 4.9E-14
HP-223	July 2 - July 9, 1980	< 5.1E-14	< 5.1E-14
HP-224	July 9 - July 16, 1980	< 4.8E-14	< 4.8E-14
HP-225	July 16 - July 23, 1980	< 5.0E-14	< 5.0E-14
HP-226	July 23 - July 30, 1980	< 4.8E-14	< 4.8E-14
HP-227	July 30 - August 6, 1980	< 6.9E-14	< 6.9E-14

The enacted Met-Ed/GPU off-site radiation monitoring program is described and then a summary of the resulting data, analyses, and findings is presented below. A more detailed examination and analysis of the data gathered is available in Reference 64.

Met-Ed/GPU manpower requirements during venting included approximately 36 people split into twelve-hour shifts. There was also substantial manpower expended in preparation for the Reactor Building venting and to analyze and report REMP results following the venting. In addition, several hundreds of thousands of dollars were spent on radiological environmental monitoring equipment acquired for the venting of Kr-85 particularly the mobile radiation environmental laboratory.

4.4.1.1 REMP Description. The Met-Ed/GPU off-site radiation monitoring program can be divided into two parts. The first part was the deployment of fixed monitoring stations. The second was the development and use of mobile monitoring capabilities to track the plume in the environment.

Fixed Monitoring. The fixed monitoring stations important to the Reactor Building purge program were composed of the following radiation monitoring and sampling devices:

1. Thermoluminescent Dosimeters (TLDs)
2. Continuous Air Samplers for Particulates and Iodines
3. Real-time Environmental Radiation Monitors
4. Continuous Noble Gas Air Samplers

Three TLD systems were deployed during the venting. Teledyne TLD badges, sensitive to penetrating radiation, were placed at 20 stations where they were changed monthly and 53 stations where they were changed quarterly. Radiation Management Corporation TLD badges, also sensitive to penetrating radiation, were located at ten of the Teledyne TLD stations abd changed monthly. Panasonic TLD badges (Model 801), sensitive to both penetrating and nonpenetrating radiation, were located at all Teledyne stations (73) and at 30 additional special stations. Panasonic badges were exposed only during the venting period. The TLD locations were chosen based on population and meteorological parameters.

Eight continuous air samplers were deployed around TMI. Table 29 provides the locations of these continuous air samplers. These samplers passed air through a particulate filter and charcoal cartridge which were then sent to a commercial laboratory for analysis. Particulate and iodine levels down to 1 E-14 μCi/cc and 7 E-14 μCi/cc respectively could be detected.

Ten stationary direct radiation monitors (Reuter-Stokes, RSS-III) were deployed and their locations are listed in Table 30. These instruments display, on a real-time basis, the gamma radiation level via an LED readout, and record the data on a stripchart and a magnetic tape. These instruments are sensitive to background radiation levels of μR per hour.

Nine cryogenic continuous air samplers were employed. The locations of these samplers are given in Table 31 and were selected based on historical meteorology and local demography. The cryogenic air samplers were set to collect ambient air continuously over a one-week period. The samples collected were analyzed at a commercial laboratory. The limits of detection (LLD) of the Kr-85 analysis is currently under study. Independent measurements, however, support the accuracy of the analysis in the range of 10^{-7} to 10^{-9} μCi/cc, Kr-85.

The fixed monitoring stations' data were collected on the following frequencies:

Monitor	Frequency
TLDs	As Stated
Continuous Air Samplers for Particulates	Weekly
Environmental Radiation Monitors	Daily
Continuous Air Samplers - Krypton 85	Weekly

Mobile Monitoring. Two mobile monitoring teams and a mobile radiation environmental laboratory were utilized by Met-Ed/GPU during the venting. The mobile monitoring teams had portable Eberline Model E-250 and Ludlum Model 177 GM survey meters equipped with pancake probes to measure Kr-85 beta dose rates. These instruments were calibrated by exposure to known concentrations of Kr-85 and had an estimated lower limit of detection (LLD) of 10^{-6} μCi/cc, Kr-85. The mobile teams also were equipped with real-time direct radiation monitors (Reuter-Stokes, RSS-III, see above) for use if plume radiocontaminiation was suspected.

TABLE 29. REMP CONTINUOUS AIR SAMPLING STATIONS

Station Code	Location	From TMI Azimuth	From TMI Distance
1S2	North Weather Station (TMI)	0^{o}	0.6 km
5A1	Observation Center	100	0.6
12B1	Goldsboro	253	2.1
1C1	Middletown	355	4.2
8C1	Falmouth	159	3.7
7F1	Marietta	127	15.8
9G1	York	180	20.3
15G1	West Fairview	306	21.6

TABLE 30. REMP STATIONARY DIRECT RADIATION MONITOR LOCATIONS

Station Code	Location	From TMI Azimuth	From TMI Distance
1C1	Middletown	355°	4.2 km
2A1	North guard shack (TMI)	23	0.6
9B1	TMI-south end of the island	160	2.4
8C1	Falmouth	159	3.7
7F1	Marietta	127	15.8
6D1	Longenecker's Farm	109	5.6
13B2	Goldsboro Marina	265	1.9
5A1	Observation Center	100	0.6
15D1	Harrisburg Airport	324	5.6
14S2	East Shelley Island	293	0.6

TABLE 31. LOCATIONS OF THE REMP STATIONARY CONTINUOUS CRYOGENIC AIR SAMPLERS

Station Code	Description	From TMI Azimuth	From TMI Distance
1S2	North Weather Station (TMI)	0^{o}	0.6 km
5A1	Observation Center	100	0.6
12B1	Goldsboro	253	2.1
1C1	Middletown	355	4.2
8C1	Falmouth	159	3.7
*15D1	Harrisburg Airport	324	5.6
7F1	Marietta	127	15.8
6G4	West Donegal	112	19.0
9G1	York	180	20.3
---	Spare Unit		

*Out of service during purge

A mobile radiation environmental laboratory was also built for the Reactor Building purge. This lab had a thirty-three foot telescoping meteorological tower that recorded wind speed, wind direction, and ambient air temperatures. This data when used in the field was compared to the on-site meteorological tower information. The mobile laboratory measured beta dose rates with a Kimmel air sampling proportional counter and the GM survey instruments described above. The proportional counter was calibrated with Kr-85 and had an estimated LLD of 3 E-8 μCi/cc. The laboratory was also equipped with the following gamma detectors: one MAB-604 plastic scintillation detector with beta shield and two Reuter-Stokes Model RSS-111 environmental monitors.

The mobile monitoring teams and the mobile lab were directed by radio to off-site locations by the Environmental Assessment Command Center (EACC). The EACC, based on meteorological information from the on-site tower, and the MIDAS computer programs positioned the mobile units where vented material was considered likely to touchdown.

4.4.1.2 REMP Results. The measurements taken of off-site dose rates by the mobile monitoring teams during the 14 day venting period were almost all consistent with normal background radiation levels. The highest dose rate (15 min at 1 mrem/hr) was measured at the TMI Observation Center on July 3, 1980. While measured dose rates on July 3, 1980 were well below the NRC venting guideline of 3 mrem/hr, off-site beta-skin dose, the venting release rate was nonetheless lowered, on the recommendation of the REMP supervisor, for much of the day. Positive dose rate measurements were detected close to TMI from June 28 to July 9, and decreased rapidly with increasing distance from TMI.

The cryogenic air sampling results gave beta-skin dose estimates ranging from 0.03 mrem (Goldsboro) to 1.8 mrem (TMI Observation Center). Gamma doses were lower than the respective beta-skin doses by a factor of 83. These doses are well below the NRC venting guidelines of 15 mrem beta-skin dose and 5 mrem gamma whole body dose.

The Panasonic TLD measurements showed only two stations which recorded statistically significant doses above the TLD's theoretical LLD for nonpenetrating radiation (LLD equals 5.3 mrem, beta-skin dose from Kr-85). The two stations recorded 7.3 $\pm$ 0.0 mrem and 8.1 $\pm$ 7.3 mrem, both of which include a natural background component estimated at two to four mrem. No penetrating radiation exposures attributable to the vented Kr-85 were recorded by the Pancsonic TLD's.

Table 32 shows the computer projections of the maximum integrated doses in each sector from the vented Kr-85. The projected beta-skin doses ranged from 0.11 mrem (sector 11) to 4.5 mrem (sector 6). Projected gamma doses ranges from 0.0027 mrem (sector 12) to 0.045 mrem (sector 6).

The average background radiation levels recorded by the Reuter-Stokes environmental monitors during venting ranged from 6 to 12 μR/hr depending on station location. Of the 35 peaks greater than 2 μR/hr above background recorded during the venting period, only seven recorded on July 1, 3, and 4 at two stations relatively close to TMI (TMI Observation Center and East Shelley Island) are thought to have a possible connection with TMI venting. These peaks were similar in size to the natural background peaks caused by local precipitation.

The results of analyses of the air particulate samples show gross beta activities similar to levels recorded before venting. The positive values indicated by the gamma scans are consistent with natural radionuclide levels. Similarly, analyses of the air iodine samples were all less than LLD, consistent with normal background levels.

The results of the monthly Teledyne TLD data showed gamma levels several mR higher than those recorded earlier in the year. This increase is believed due to the fact that no compensation was made for transit exposures in the June and July Teledyne measurements. Transit exposures (typically 1 to 3 mR) are normally subtracted from the Teledyne gross exposure measurements. Difficulties with the transit of badges in June and July, however, led to inaccurate transit exposure estimates which were not subtracted from the badge readings. This conclusion is supported by the fact that neither of the other types of TLD badges showed increased gamma levels during June and July. Quarterly TLD data also showed no increase in gamma levels during the venting interval.

The Met-Ed/GPU environmental radiation monitoring results therefore support the conclusions that:

TABLE 32. MIDAS COMPUTER PROJECTIONS OF RADIATION DOSES FROM VENTED KRYPTON-85

Sector[a]	Direction From TMI	Maximum Projected Beta-Skin Dose		Maximum Projected Gamma-Whole Body Dose	
1	N	1.8 E 0 mrem	@ 2.0 km	2.7 E-2 mrem	@ 0.6 km
2	NNE	2.6 E 0	0.6	3.5 E-2	0.6
3	NE	1.8 E 0	0.6	2.7 E-2	0.6
4	ENE	1.7 E 0	1.5	1.5 E-2	0.6
5	E	2.3 E 0	1.0	2.7 E-2	0.6
6	ESE	4.5 E 0	0.6	4.5 E-2	0.6
7	SE	2.3 E 0	0.6	2.3 E-2	0.6
8	SSE	1.9 E 0	0.6	1.9 E-2	0.6
9	S	1.5 E 0	0.6	2.6 E-2	0.6
10	SSW	9.7 E-1	0.6	1.1 E-2	0.6
11	SW	1.1 E-1	1.0	3.6 E-3	0.6
12	WSW	2.3 E-1	2.0	2.7 E-3	0.6
13	W	6.2 E-1	0.6	1.1 E-2	0.6
14	WNW	4.0 E-1	0.6	4.3 E-3	0.6
15	NW	8.4 E-1	2.0	1.2 E-3	0.6
16	NNW	5.4 E-1	2.0	5.8 E-3	0.6

a) Each sector originates at TMI and extends radially in the direction indicated.

1. Off-site dose rates from the vented material did not exceed the NRC venting guidelines of 3 mrem/hr, beta skin dose, and 1 mrem/hr whole body dose.

2. Off-site integrated doses from the vented material did not exceed NRC guidelines of 15 mrem, beta-skin dose and 5 mrem, whole body dose. Kr-85 dose estimates suggest the maximum off-site doses were 2 to 5 mrem, beta-skin dose and less than 1 mrem, whole body dose.

3. No significant amounts of plume radiocontaminants were detected off-site.

4.4.2 On-Site Radiation Monitoring

During the venting of Kr-85 from the TMI-2 Reactor Building, radiation monitoring of the on-site (owner-controlled) area of TMI was conducted under the direction of the Met-Ed Radiological Technical Support (RTS) Group. This monitoring was intended to assure that no unexpected exposures to individuals occurred at the TMI site. A temporary shutdown of the purge was to occur if a dose rate equivalent to 10 mR/hr (for a skin dose exposure rate) were determined to exist to an individual outside the "protected area" (individual assumed not to be wearing a TLD badge) or if any unusual or unexplained dose rates were measured.

The on-site monitoring activities consisted of measuring radiation levels at designated locations all over the TMI site where background radiation levels had previously been established. Measurements were made with three specially calibrated radiation monitoring instruments.

- HPI model 1072 air equivalent, unsealed ionization chamber for gross gamma radiation detection (reading in mR/hr gamma).
- Ludlum model 16 analyzer with a 1" x 1" NAI scintillation detector for gamma radiation detection (reading in cpm).
- Eberline E140(N) with an HP260, thin window GM detector for beta detection (reading in cpm)

Each of these instruments were calibrated for a specific function. The HPI 1072 was calibrated to Cs-137 from 0.15 mR/hr to 200 mR/hr. It was used for gamma radiation only and is not beta sensitive. The Ludlum 16 was calibrated as a single channel analyzer with the peak set to provide maximum sensitivity to the Kr-85 gamma photon yet keep background levels as low as possible. The Eberline E140(N) with HP260 thin window pancake detector was calibrated to Kr-85 beta and had the highest sensitivity to Kr-85.

Starting with the initial venting activities on June 28 and until the afternoon of July 1, on-site surveys were conducted at least once per hour, 24 hours per day while venting. During this time period, an RTS coordinator was located in the Unit #2 Control Room to direct two survey personnel by way of two-way radios on where to take on-site radiation measurements based on meteorological tower recorder output, purge rate, and stack exhaust rate. All radiation measurements were recorded on a record sheet like the one shown in Table 33 by the survey personnel. The survey personnel also called in the radiation readings to the RTS coordinator who kept a similar record. The initial intensity of monitoring was designed to characterize radiological on-site parameters particularly in terms of the two variables, (1) Reactor Building purge rate and (2) TMI site meteorology. Starting the evening of July 1, the on-site monitoring program was relaxed and downwind survey measurements were made only every four hours. For these four hour surveys, the survey personnel contacted the Control Room to obtain wind direction and then surveyed the on-site area downwind of the station vent. They simply recorded their measured results unless any abnormally high readings were observed in which case the Control Room was notified.

During the 14 day purge, approximately 4500 field readings were taken. No gamma radiation levels above background were measured with the HPI 1072. The highest readings seen on the other two instruments occurred during the first five days of the purge and at the beginning of the "fast" purge (July 8, 1980). These highest readings corresponded to transition meteorological conditions generally occurring during morning and evening hours. Of the approximately 1500 radiation readings recorded with the E140 (N), only 67 (approximately 4.5%) were greater than a skin does rate equivalent to 0.3 mR/hr. Of these 67 readings, four were recoreded on the roof of the Unit II Turbine Building and only three (one on the Turbine Building) were greater than an equivalent skin does of 1.9 mR/hr. The highest reading, corresponding to a skin dose of 2.6 mR/hr, was recorded on July 8, 1980 at the east side of the Unit II "protected area" when the purge commenced with the larger purge system at a purge rate of 1000 cfm.[65]

TABLE 33

ON-SITE MONITORING RESULTS DURING CONTROLLED R/B PURGE

Date	Time	Location	Instrument ______ Number ______ Cal. Date ______	Instrument ______ Number ______ Cal. Date ______	Instrument ______ Number ______ Cal. Date ______	Comments	Name

4.4.3 Auxiliary Building Radiation Monitoring

To insure the prompt detection of leakage from the Modified Hydrogen Control System (MHCS) or the "B" Train of the Modified Reactor Building Purge System (MRBPS), radiation monitoring of the Auxiliary Building was conducted at specific locations in the vicinity of these systems during their operation. This monitoring was in addition to area radiation monitor HPR-3236 located near the MHCS filter train and exhaust fan (see Section 3.2.4.3).

The monitoring initially consisted of one AMS-3 located inside the fan room contaning the MHCS and MRBPS and one located just outside the fan room door. The AMS-3 located outside the fan room had been modified and especially calibrated for Kr-85 (140 cpm corresponded to 1.0 E-5 μCi/cc). Radiation readings were initially recorded about every 20 minutes when the purge was in progress for both AMS-3's, for the control point outside the fan room measured with an RM-14, and for various points within the fan room measured with a portable survey instrument (RO2), for both beta and gamma (closed and open window respectively). Marinelli gas samples were also taken periodically in the fan room and analyzed. These readings showed that both the MHCS and the MRBPS had leaks that caused Kr-85 concentrations to reach significant levels within the fan room. For the MHCS, the leaks were small enough that the fan room concentrations did not build up too high; e.g., the fan room ventilation was sufficient to evacuate the gas as it leaked into the fan room. When the MRBPS began operation, however, the Kr-85 concentration built up in the fan room within five hours to a peak of approximately 186 times the maximum permissible concentration (MPC) level or about 1.86 E-3 μCi/cc as measured by the AMS-3 in the fan room.

The buildup of Kr-85 in the fan room was sufficient to cause the fan room radiation monitors to alarm. Kr-85 also leaked from the fan room in sufficient quantities to cause the RM-14 and the AMS-3 outside the fan room to reach their alarm points. In addition, because of unusual weather conditions, enough released Kr-85 was drawn back into the building air intake to cause a high alarm on the Control Room intake monitor (HPR-220) and the RM-14 located int he shift supervisor's office.

During this time of high Kr-85 levels, the AMS-3 monitor located outside the fan room, which was then reading 3000 cpm, was moved into the fan room and almost immediately increased its reading by a factor of 3. Several minutes later the filter was removed from this AMS-3 and replaced with a clear one. At the same time a Marinelli gas sample (1640 cc) was taken inside the room. The removed filter was surveyed with an RM-14 and found to be "clean" indicating that the results of the Marinelli sample showed a Kr-85 particulate acitivity was not being released. This was later confirmed by spectral analysis of the filter which showed no detectable isotope concentration of 3.54 E-4 μCi/cc inside the fan room.[65]

The Kr-85 concentration was eventually reduced by locating and repairing leaks in two ventilation system penthouse penetrations and in he doors leading into the penthouse. The MHCS was also run in a configuration to exhaust air from the fan room out the vent stack. It is interesting to note that investigation of leaks showed that Kr-85 diffused out of openings in ductwork even against an established pressure gradient.

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APPENDIX A
HOURLY RECORD OF Kr-85 RELEASES

Table A-1 provides an hourly record of the Kr-85 released during the venting of the Reactor Building between June 28, 1980 and July 11, 1980 as monitored by HPR-219A. Table A-2 is a summary of the daily totals for Kr-85 curies released. Included in Table A-1 for every hour are station vent flow rate, measured gaseous (Kr-85) concentration, Kr-85 release rate, and total curies of Kr-85 released. Both tables are based on work done by Porter Consultants, Inc. The total curies of Kr-85 released reported in the tables differ slightly from the final officially reported Met-Ed/GPU numbers discussed in Section 3.7 because of the different correction factors utilized by Porter Consultants, Inc. A brief discussion of the correction factors utilized by Porter Consultants is provided below.

As discussed in Section 3.7 to obtain a better estimate of the actual number of curies of Kr-85 released, errors in the measured station vent flow rate and station vent Kr-85 concentration measured by HPR-219A were required. The correction factor for flowrate used by Porter Consultants in Table A-1 was a 6.1% increase in the measured flowrate. This was determined from the difference between the data from the detailed traverse of the stack (37 points) made on July 7, 1980 and the flow as determined from the velocity recorder reading (116,195 cfm versus 109,155 cfm).

Two correction factors to HPR-219A readings of Kr-85 concentration in the station vent were used by Porter Consultants. The first one was a background correction factor of 91 cpm or approximately 3.5 E-6 μCi/cc which reduces all HPR-219A readings. This background error was discovered after the venting had been completed when HPR-219A still showed higher than background levels of radioactive material being released in the normal Auxiliary and Fuel Handling Building ventilation exhausts. The needed correction factor was determined by passing pure nitrogen through HPR-219A.

The other HPR-219A correction factor was to correct for the 3 psi pressure difference between the stack and the stack sampling line at HPR-219A. This correction factor of 1.26 increased all HPR-219A readings.

All three of the above correction factors were used to arrive at the Kr-85 release data provided in Tables A-1 and A-2. The total curies of Kr-85 released computed in this manner by Porter Consultants is 46,094 curies versus the official Met-Ed/GPU computed range from 38,302 to 50,254 curies with a median value of 44,132 curies.

TABLE A-1. Kr-85 RELEASES DURING REACTOR BUILDING VENTING JUNE 28 to JULY 11, 1980

& Time Fr	To	Cubic Feet/ Minute	cc/second	μCi/cc	μCi/second	Sample Time/ Second	Total Curies	Curies (Running Total)
6/28 @ 0700	6/28 @ 0800	1.11E5	5.23E7	4.16E-7	2.18E1	3.6E3	7.84E-2	7.84E-2
0800	0900	1.11E5	5.23E7	7.31E-7	3.83E1	3.6E3	1.38E-1	2.16E-1
1700	1800	1.08E5	5.11E7	2.28E-4	1.17E4	3.6E3	4.19E1	4.22E1
1800	1900	1.08E5	5.11E7	6.29E-4	3.21E4	3.6E3	1.16E2	1.58E2
1900	2000	1.14E5	5.36E7	6.08E-5	3.26E3	3.6E3	1.17E1	1.70E2
2000	2100	1.14E5	5.36E7	3.82E-4	2.05E4	3.6E3	7.37E1	2.43E2
2100	2200	1.14E5	5.36E7	4.93E-4	2.64E4	3.6E3	9.50E1	3.38E2
2200	2300	1.14E5	5.36E7	6.52E-5	3.49E2	3.6E3	1.26E1	3.51E2
							Daily Total	351 Ci

TABLE A-I (cont'd)

Date & Time From	To	Cubic Feet/ Minute	cc/second	μCi/cc	μCi/second	Sample Time/ Second	Total Curies	Curies (Running Total)
6/29 @ 1400	6/29 @ 1500	1.08E5	5.11E7	8.63E-4	4.41E4	3.6E3	1.59E2	1.59E2
1500	1600	1.08E5	5.11E7	1.31E-3	6.68E4	3.6E3	2.41E2	3.99E2
1600	1700	1.10E5	5.17E7	6.34E-4	3.28E4	3.6E3	1.18E2	5.17E2
1700	1800	1.12E5	5.30E7	5.61E-4	2.97E4	3.6E3	1.07E2	6.24E2
1800	1900	1.12E5	5.30E7	5.37E-4	2.84E4	3.6E3	1.02E2	7.27E2
1900	2000	1.12E5	5.30E7	7.53E-4	3.99E4	3.6E3	1.44E2	8.70E2
2000	2100	1.12E5	5.30E7	7.08E-4	3.75E4	3.6E3	1.35E2	1.01E3
2100	2200	1.18E5	5.55E7	4.37E-4	2.42E4	3.6E3	8.73E1	1.09E3
2200	2300	1.16E5	5.48E7	5.63E-4	3.09E4	3.6E3	1.11E2	1.20E3
2300	2400	1.15E5	5.42E7	6.46E-4	3.50E4	3.6E3	1.26E2	1.33E3
							Daily Total	1,330 Ci

TABLE A-1 (cont'd)

Date & Time From	To	Cubic Feet/ Minute	cc/second	µCi/cc	µCi/second	Sample Time/ Second	Total Curies	Curies (Running Total)
6/30 @ 0000	6/30 @ 0100	1.15E5	5.42E7	5.78E-4	3.14E4	3.6E3	1.13E2	1.13E2
0100	0200	1.15E5	5.42E7	5.00E-4	2.71E4	3.6E3	9.76E1	2.11E2
0200	0300	1.15E5	5.42E7	8.81E-6	4.78E2	3.6E3	1.72E0	2.12E2
0300	0400	1.15E5	5.42E7	6.75E-5	3.66E3	3.6E3	1.32E1	2.25E2
0400	0500	1.15E5	5.42E7	5.98E-4	3.24E4	3.6E3	1.17E2	3.42E2
0500	0600	1.15E5	5.42E7	7.81E-4	4.24E4	3.6E3	1.52E2	4.95E2
0600	0700	1.15E5	5.42E7	6.36E-4	3.45E4	3.6E3	1.24E2	6.19E2
0700	0800	1.15E5	5.42E7	7.26E-4	3.93E4	3.6E3	1.42E2	7.61E2
0800	0900	1.15E5	5.42E7	1.06E-3	5.76E4	3.6E3	2.07E2	9.68E2
0900	1000	1.15E5	5.42E7	1.04E-3	5.63E4	3.6E3	2.03E2	1.17E3
1000	1100	1.15E5	5.42E7	1.05E-3	5.68E4	3.6E3	2.04E2	1.37E3
1100	1200	1.14E5	5.36E7	1.01E-3	5.44E4	3.6E3	1.96E2	1.57E3
1200	1300	1.14E5	5.36E7	6.30E-4	3.38E4	3.6E3	1.22E2	1.69E3
1300	1400	1.14E5	5.36E7	1.06E-3	5.69E4	3.6E3	2.05E2	1.90E3
1400	1500	1.14E5	5.36E7	1.09E-3	5.83E4	3.6E3	2.10E2	2.11E3
1500	1600	1.12E5	5.30E7	1.12E-3	5.93E4	3.6E3	2.13E2	2.32E3
1600	1700	1.12E5	5.30E7	9.27E-4	4.91E4	3.6E3	1.77E2	2.50E3

TABLE A-1 (cont'd)

Date & Time From	To	Cubic Feet/ Minute	cc/second	μCi/cc	μCi/second	Sample Time/ Second	Total Curies	Curies (Running Total)
6/30 @ 1700	6/30 @ 1800	1.12E5	5.30E7	1.28E-3	6.79E4	3.6E3	2.45E2	2.74E3
1800	1900	1.14E5	5.36E7	1.30E-3	6.94E4	3.6E3	2.50E2	2.99E3
1900	2000	1.14E5	5.36E7	1.70E-3	9.10E4	3.6E3	3.28E2	3.32E3
2000	2100	1.14E5	5.36E7	8.69E-4	4.66E4	3.6E3	1.68E2	3.49E3
2100	2200	1.14E5	5.36E7	7.04E-4	3.77E4	3.6E3	1.36E2	3.62E3
2200	2300	1.15E5	5.42E7	1.36E-3	7.36E4	3.6E3	2.65E2	3.89E3
2300	2400	1.15E5	5.42E7	1.30E-3	7.02E4	3.6E3	2.53E2	4.14E3
							Daily Total	4,140 Ci

TABLE A-1 (cont'd)

Date & Time From	To	Cubic Feet/ Minute	cc/second	μCi/cc	μCi/second	Sample Time/ Second	Total Curies	Curies (Running Total)
7/1 @ 0000	7/1 @ 0100	1.15E5	5.42E7	1.10E-3	5.99E4	3.6E3	2.16E2	2.16E2
0100	0200	1.16E5	5.48E7	1.10E-3	6.05E4	3.6E3	2.18E2	4.34E2
0200	0300	1.16E5	5.48E7	1.03E-3	5.67E4	3.6E3	2.04E2	6.38E2
0300	0400	1.18E5	5.55E7	1.14E-3	6.32E4	3.6E3	2.28E2	8.65E2
0400	0500	1.13E5	5.55E7	1.09E-3	6.05E4	3.6E3	2.18E2	1.08E3
0500	0600	1.18E5	5.55E7	1.15E-3	6.39E4	3.6E3	2.30E2	1.31E3
0600	0700	1.18E5	5.55E7	8.87E-4	4.92E4	3.6E3	1.77E2	1.49E3
0700	0800	1.16E5	5.48E7	9.35E-4	5.13E4	3.6E3	1.85E2	1.67E3
0800	0900	1.16E5	5.48E7	2.39E-3	1.31E5	3.6E3	4.72E2	2.15E3
0900	1000	1.16E5	5.48E7	1.55E-3	8.49E4	3.6E3	3.05E2	2.45E3
1000	1100	1.14E5	5.36E7	2.87E-3	1.54E5	3.6E3	5.54E2	3.01E3
1100	1200	1.11E5	5.23E7	1.48E-3	7.77E4	3.6E3	2.80E2	3.29E3
1200	1300	1.08E5	5.11E7	8.11E-5	4.14E3	3.6E3	1.49E1	3.30E3
1300	1400	1.07E5	5.05E7	9.17E-4	4.63E4	3.6E3	1.67E2	3.47E3
1400	1500	1.07E5	5.05E7	1.22E-3	6.15E4	3.6E3	2.21E2	3.69E3
1500	1600	1.07E5	5.05E7	1.51E-3	7.67E4	3.6E3	2.74E2	3.96E3
1600	1700	1.11E5	5.23E7	1.96E-3	1.03E5	3.6E3	3.70E2	4.33E3

TABLE A-I (cont'd)

Date & Time		Cubic Feet/ Minute	cc/second	μCi/cc	μCi/second	Sample Time/ Second	Total Curies	Curies (Running Total)
From	To							
7/1 @ 1700	7/1 @ 1800	1.11E5	5.23E7	2.30E-3	1.21E5	3.6E3	4.34E2	4.77E3
1800	1900	1.15E5	5.23E7	2.97E-3	1.56E5	3.6E3	5.60E2	5.33E3
1900	2000	1.12E5	5.30E7	2.28E-3	1.21E5	3.6E3	4.34E2	5.76E3
2000	2100	1.14E5	5.36E7	1.46E-3	7.82E4	3.6E3	2.82E2	6.04E3
2100	2200	1.14E5	5.36E7	1.18E-3	6.34E4	3.6E3	2.28E2	6.27E3
2200	2300	1.14E5	5.36E7	9.75E-3	5.23E4	3.6E3	1.88E2	6.46E3
2300	2400	1.16E5	5.48E7	1.15E-3	6.32E4	3.6E3	2.28E2	6.69E3
							Daily Total	6,690 Ci

TABLE A-1 (cont'd)

Date & Time		Cubic Feet/ Minute	cc/second	µCi/cc	µCi/second	Sample Time/ Second	Total Curies	Curies (Running Total)
From	To							
7/2 @ 0000	7/2 @ 0013	1.16E5	5.48E7	1.37E-3	7.52E4	3.6E3	5.86E1	5.86E1
0013	0300	Computer Down						
0300	0400	1.16E5	5.48E7	3.96E-5	2.17E3	3.6E3	7.82E0	6.65E1
0400	0500	1.16E5	5.48E7	8.45E-6	4.64E2	3.6E3	1.67E0	6.81E1
0500	0600	1.16E5	5.48E7	7.07E-4	3.88E4	3.6E3	1.40E2	2.077E2
0600	0700	1.16E5	5.48E7	1.40E-3	7.66E4	3.6E3	2.76E2	4.83E2
0700	0800	1.16E5	5.48E7	1.32E-3	7.24E4	3.6E3	2.61E2	7.44E2
0800	0900	1.15E5	5.42E7	1.37E-3	7.43E4	3.6E3	2.68E2	1.01E3
0900	1000	1.14E5	5.36E7	2.08E-3	1.11E5	3.6E3	4.01E2	1.41E3
1000	1100	1.12E5	5.30E7	2.77E-3	1.47E5	3.6E3	5.28E2	1.94E3
1100	1200	1.10E5	5.17E7	2.87E-3	1.48E5	3.6E3	5.34E2	2.47E3
1200	1300	1.08E5	5.11E7	1.70E-4	8.69E3	3.6E3	3.13E1	2.51E3
1300	1400	1.07E5	5.05E7	2.03E-5	1.03E3	3.6E3	3.70E0	2.51E3
1400	1500	1.07E5	5.05E7	1.49E-5	7.54E2	3.6E3	2.71E0	2.52E3
1500	1600	1.07E5	5.05E7	1.50E-3	7.56E4	3.6E3	2.72E2	2.78E3
1600	1700	1.08E5	5.11E7	2.22E-3	1.13E5	3.6E3	4.07E2	3.19E3
1700	1800	1.08E5	5.11E7	2.09E-3	1.07E5	3.6E3	3.84E2	3.58E3

TABLE A-1 (cont'd)

Date & Time From	To	Cubic Feet/ Minute	cc/second	µCi/cc	µCi/second	Sample Time/ Second	Total Curies	Curies (Running Total)
7/2 @ 1800	7/2 @ 1900	1.08E5	5.11E7	1.91E-3	9.77E4	3.6E3	3.52E2	3.93E3
1900	2000	1.08E5	5.11E7	1.53E-3	7.84E4	3.6E3	2.82E2	4.21E3
2000	2100	1.10E5	5.17E7	1.32E-3	6.83E4	3.6E3	2.46E2	4.46E3
2100	2200	1.10E5	5.17E7	1.12E-3	5.79E4	3.6E3	2.08E2	4.66E3
2200	2300	1.10E5	5.17E7	7.67E-4	3.97E4	3.6E3	1.43E2	4.81E3
2300	2400	1.11E5	5.23E7	7.67E-4	4.02E4	3.6E3	1.45E2	4.95E3
							Daily Total	4,950 Ci

TABLE A-I (cont'd)

Date & Time From	To	Cubic Feet/ Minute	cc/second	μCi/cc	μCi/second	Sample Time/ Second	Total Curies	Curies (Running Total)
7/3 @ 0000	7/3 @ 0100	1.14E5	5.36E7	9.06E-4	4.86E4	3.6E3	1.75E2	1.75E2
0100	0200	1.14E5	5.36E7	1.38E-3	7.41E4	3.6E3	2.67E2	4.42E2
0200	0300	1.14E5	5.36E7	2.22E-3	1.19E5	3.6E3	4.27E2	8.69E2
0300	0400	1.15E5	5.42E7	1.90E-3	1.03E5	3.6E3	3.71E2	1.24E3
0400	0500	1.15E5	5.42E7	1.57E-3	8.53E4	3.6E3	3.07E2	1.55E3
0500	0600	1.14E5	5.36E7	1.30E-3	6.94E4	3.6E3	2.50E2	1.80E3
0600	0700	1.14E5	5.36E7	2.39E-5	1.28E3	3.6E3	4.61E0	1.80E3
0700	0800	1.14E5	5.36E7	1.38E-5	7.39E2	3.6E3	2.66E0	1.80E3
0800	0900	1.14E5	5.36E7	6.90E-4	3.70E4	3.6E3	1.33E2	1.94E3
0900	1000	1.15E5	5.42E7	1.36E-3	7.36E4	3.6E3	2.65E2	2.20E3
1000	1100	1.14E5	5.36E7	1.86E-3	9.98E4	3.6E3	3.59E2	2.56E3
1100	1200	1.14E5	5.36E7	1.45E-3	7.75E4	3.6E3	2.79E2	2.84E3
1200	1300	1.15E5	5.42E7	1.11E-3	6.01E4	3.6E3	2.16E2	3.06E3
1300	1400	1.15E5	5.42E7	1.26E-3	6.82E4	3.6E3	2.45E2	3.30E3
1400	1500	1.12E5	5.30E7	5.90E-4	3.12E4	3.6E3	1.12E2	3.42E3
1500	1600	1.12E5	5.30E7	1.01E-3	5.35E4	3.6E3	1.92E2	3.61E3
1600	1700	1.11E5	5.23E7	1.10E-3	5.76E4	3.6E3	2.07E2	3.81E3

TABLE A-1 (cont'd)

Date & Time From	To	Cubic Feet/ Minute	cc/second	μCi/cc	μCi/second	Sample Time/ Second	Total Curies	Curies (Running Total)
7/3 @ 1700	7/3 @ 1800	1.10E5	5.17E7	1.95E-3	1.01E5	3.6E3	3.63E2	4.18E3
1800	1900	1.10E5	5.17E7	2.09E-3	1.08E5	3.6E3	3.89E2	4.57E3
1900	2000	1.11E5	5.23E7	1.56E-3	8.17E4	3.6E3	2.94E2	4.86E3
2000	2100	1.11E5	5.23E7	8.34E-4	4.37E4	3.6E3	1.57E2	5.02E3
2100	2200	1.11E5	5.23E7	1.32E-3	6.91E4	3.6E3	2.49E2	5.27E3
2200	2300	1.14E5	5.36E7	1.28E-3	6.87E4	3.6E3	2.47E2	5.51E3
2300	2400	1.14E5	5.36E7	1.43E-3	7.68E4	3.6E3	2.77E2	5.79E3
							Daily Total	5,790 Ci

TABLE A-1 (cont'd)

Date & Time From	To	Cubic Feet/ Minute	cc/second	µCi/cc	µCi/second	Sample Time/ Second	Total Curies	Curies (Running Total)
7/4 @ 0000	7/4 @ 0100	1.14E5	5.36E7	1.14E-3	6.09E4	3.6E3	2.19E2	3.19E2
0100	0200	1.15E5	5.42E7	1.50E-3	8.12E4	3.6E3	2.92E2	5.11E2
0200	0300	1.15E5	5.42E7	1.56E-3	8.46E4	3.6E3	3.04E2	8.16E2
0300	0400	1.15E5	5.42E7	1.53E-3	8.32E4	3.6E3	3.00E2	1.12E3
0400	0500	1.15E5	5.42E7	1.52E-3	8.25E4	3.6E3	2.97E2	1.41E3
0500	0600	1.14E5	5.36E7	1.46E-3	7.82E4	3.6E3	2.82E2	1.69E3
0600	0700	1.14E5	5.36E7	1.31E-3	7.01E4	3.6E3	2.52E2	1.95E3
0700	0800	1.14E5	5.36E7	9.71E-4	5.21E4	3.6E3	1.87E2	2.134E3
0800	0900	1.14E5	5.36E7	1.22E-3	6.54E4	3.6E3	2.36E2	2.37E3
0900	1000	1.14E5	5.36E7	1.61E-3	8.63E4	3.6E3	3.11E2	2.68E2
1000	1100	1.12E5	5.30E7	1.64E-3	8.66E4	3.6E3	3.12E2	2.99E3
1100	1200	1.12E5	5.30E7	1.64E-3	8.66E4	3.6E3	3.12E2	3.30E3
1200	1300	1.07E5	5.05E7	1.61E-3	8.13E4	3.6E3	2.93E2	3.60E3
1300	1400	1.06E5	4.99E7	1.59E-3	7.90E4	3.6E3	2.84E2	3.88E3
1400	1500	1.06E5	4.99E7	1.55E-3	7.71E4	3.6E3	2.78E2	4.16E3
1500	1600	1.06E5	4.99E7	1.52E-3	7.59E4	3.6E3	2.73E2	4.43E3
1600	1700	1.08E5	5.11E7	1.42E-3	7.26E4	3.6E3	2.61E2	4.69E3

TABLE A-1 (cont'd)

Date & Time From	Date & Time To	Cubic Feet/ Minute	cc/second	μCi/cc	μCi/second	Sample Time/ Second	Total Curies	Curies (Running Total)
7/4 @ 1700	7/4 @ 1800	1.07E5	5.05E7	1.31E-3	6.60E4	3.6E3	2.38E2	4.93E3
1800	1900	1.08E5	5.11E7	1.45E-3	7.39E4	3.6E3	2.66E2	5.20E3
1900	2000	1.08E5	5.11E7	1.28E-3	6.55E4	3.6E3	2.36E2	5.4[illegible]E3
2000	2100	1.10E5	5.17E7	8.29E-4	4.29E4	3.6E3	1.54E2	5.59E3
2100	2200	1.14E5	5.36E7	5.47E-4	2.93E4	3.6E3	1.05E2	5.69E3
2200	2300	1.14E5	5.36E7	5.17E-4	2.77E4	3.6E3	9.97E0	5.79E3
2300	2400	1.14E5	5.36E7	1.28E-4	6.88E3	3.6E3	2.48E1	5.82E3
							Daily Total	5,820 Ci

TABLE A-1 (cont'd)

Date & Time From	To	Cubic Feet/ Minute	cc/second	μCi/cc	μCi/second	Sample Time/ Second	Total Curies	Curies (Running Total)
7/5 @ 0000	7/5 @ 0100	1.14E5	5.36E7	1.12E-5	5.98E2	3.6E3	2.15E0	2.15E0
0100	0200	1.15E5	5.42E7	9.60E-6	5.21E2	3.6E3	1.87E0	4.03E0
0200	0300	1.16E5	5.48E7	8.48E-6	4.65E2	3.6E3	1.67E0	5.70E0
0300	0400	1.16E5	5.48E7	3.70E-4	2.03E4	3.6E3	7.31E1	7.88E1
0400	0500	1.15E5	5.42E7	5.04E-4	2.73E4	3.6E3	9.84E1	1.77E2
0500	0600	1.15E5	5.42E7	6.31E-4	3.42E4	3.6E3	1.23E2	3.00E2
0600	0700	1.15E5	5.42E7	9.80E-4	5.31E4	3.6E3	1.91E2	4.92E2
0700	0800	1.16E5	5.48E7	1.16E-3	5.36E4	3.6E3	2.29E2	7.21E2
0800	0900	1.14E5	5.36E-3	1.19E-3	6.40E4	3.6E3	2.30E2	9.51E2
0900	1000	1.11E5	5.23E7	9.32E-4	4.88E4	3.6E3	1.76E2	1.13E3
1000	1100	1.12E5	5.30E7	5.43E-4	2.88E4	3.6E3	1.04E2	1.23E3
1100	1200	1.10E5	5.17E7	1.50E-3	7.74E4	3.6E3	2.79E2	1.51E3
1200	1300	1.06E5	4.99E7	1.46E-3	7.27E4	3.6E3	2.62E2	1.77E3
1300	1400	1.07E5	5.05E7	1.43E-3	7.24E4	3.6E3	2.61E2	2.03E3
1400	1500	1.06E5	4.99E7	1.41E-3	7.02E4	3.6E3	2.53E2	2.28E3
1500	1600	1.07E5	5.05E7	1.40E-3	7.05E4	3.6E3	2.54E2	2.54E3
1600	1700	1.07E5	5.05E7	1.36E-3	6.86E4	3.6E3	2.47E2	2.79E3

TABLE A-1 (cont'd)

Date & Time From	To	Cubic Feet/ Minute	cc/second	μCi/cc	μCi/second	Sample Time/ Second	Total Curies	Curies (Running Total)
7/5 @ 1700	7/5 @ 1800	1.06E5	4.99E7	1.32E-3	6.58E4	3.6E3	2.37E2	3.02E3
1800	1900	1.06E5	4.99E7	1.32E-3	6.58E4	3.6E3	2.37E2	3.26E3
1900	2000	1.08E5	5.11E7	9.35E-4	4.78E4	3.6E3	1.72E2	3.43E3
2000	2100	1.08E5	5.11E7	6.98E-4	3.57E4	3.6E3	1.28E2	3.56E3
2100	2200	1.10E5	5.17E7	7.63E-4	3.95E4	3.6E3	1.42E2	3.70E3
2200	2300	1.10E5	5.17E7	5.74E-4	2.97E4	3.6E3	1.07E2	3.81E3
2300	2400	1.10E5	5.17E7	6.45E-4	3.34E4	3.6E3	1.20E2	3.93E3
							Daily Total	3,930 Ci

TABLE A-I (cont'd)

Date & Time From	Date & Time To	Cubic Feet/ Minute	cc/second	μCi/cc	μCi/second	Sample Time/ Second	Total Curies	Curies (Running Total)
7/6 @ 0000	7/6 @ 0100	1.11E5	5.23E7	3.87E-4	2.02E4	3.6E3	7.29E1	7.29E1
0100	0200	1.11E5	5.23E7	3.19E-4	1.67E4	3.6E3	6.01E1	1.33E2
0200	0300	1.11E5	5.23E7	5.23E-4	2.74E4	3.6E3	9.85E1	2.31E2
0300	0400	1.11E5	5.23E7	6.35E-4	3.32E4	3.6E3	1.20E2	3.51E2
0400	0500	1.12E5	5.30E7	7.40E-4	3.92E4	3.6E3	1.41E2	4.92E2
0500	0600	1.14E5	5.36E7	7.99E-4	4.28E4	3.6E3	1.54E2	6.46E2
0600	0700	1.16E5	5.48E7	7.17E-4	3.93E4	3.6E3	1.42E2	7.88E2
0700	0800	1.16E5	5.48E7	8.68E-4	4.76E4	3.6E3	1.71E2	9.59E2
0800	0900	1.16E5	5.48E7	1.08E-3	5.90E4	3.6E3	2.12E2	1.17E3
0900	1000	1.18E5	5.55E7	1.04E-3	5.79E4	3.6E3	2.09E2	1.38E3
1000	1100	1.18E5	5.55E7	1.07E-3	5.96E4	3.6E3	2.15E2	1.59E3
1100	1200	1.16E5	5.48E7	1.04E-3	5.69E4	3.6E3	2.05E2	1.80E3
1200	1300	1.16E5	5.48E7	1.01E-3	5.53E4	3.6E3	1.99E2	1.99E3
1300	1400	1.16E5	5.48E7	9.76E-4	5.35E4	3.6E3	1.93E2	2.19E3
1400	1500	1.14E5	5.36E7	9.81E-4	5.26E4	3.6E3	1.89E2	2.38E3
1500	1600	1.14E5	5.36E7	9.55E-4	5.12E4	3.6E3	1.84E2	2.57E3
1600	1700	1.14E5	5.36E7	9.42E-4	5.05E4	3.6E3	1.82E2	2.75E3

TABLE A-I (cont'd)

Date & Time		Cubic Feet/ Minute	cc/second	μCi/cc	μCi/second	Sample Time/ Second	Total Curies	Curies (Running Total)
From	To							
7/6 @ 1700	7/6 @ 1800	1.14E5	5.36E7	9.27E-4	4.97E4	3.6E3	1.79E2	2.93E3
1800	1900	1.14E5	5.36E7	9.07E-4	4.86E4	3.6E3	1.75E2	3.10E3
1900	2000	1.14E5	5.36E7	9.02E-4	4.83E4	3.6E3	1.74E2	3.28E3
2000	2100	1.14E5	5.36E7	8.45E-4	4.53E4	3.6E3	1.63E2	3.44E3
2100	2200	1.15E5	5.42E7	6.63E-4	3.59E4	3.6E3	1.29E2	3.57E3
2200	2300	1.16E5	5.48E7	6.56E-4	3.60E4	3.6E3	1.30E2	3.70E3
2300	2400	1.16E5	5.48E7	6.05E-4	3.32E4	3.6E3	1.19E2	3.82E3
							Daily Total	3,820 Ci

TABLE A-1 (cont'd)

Date & Time		Cubic Feet/ Minute	cc/second	μCi/cc	μCi/second	Sample Time/ Second	Total Curies	Curies (Running Total)
From	To							
7/7 @ 0000	7/7 @ 0100	1.16E5	5.48E7	3.17E-4	1.74E4	3.6E3	6.27E1	6.27E1
0100	0200	1.18E5	5.55E7	2.08E-4	1.15E4	3.6E3	4.15E1	1.04E2
0200	0300	1.19E5	5.61E7	4.27E-4	2.40E4	3.6E3	8.62E1	1.90E2
0300	0400	1.19E5	5.61E7	6.59E-4	3.70E4	3.6E3	1.33E2	3.23E2
0400	0500	1.19E5	5.61E7	6.60E-4	3.70E4	3.6E3	1.33E2	4.57E2
0500	0600	1.20E5	5.67E7	3.58E-4	2.03E4	3.6E3	7.30E1	5.30E2
0600	0700	1.19E5	5.61E7	3.26E-4	1.83E4	3.6E3	6.59E1	5.96E2
0700	0800	1.19E5	5.61E7	5.01E-4	2.81E4	3.6E3	1.01E2	6.97E2
0800	0900	1.16E5	5.48E7	7.58E-4	4.16E4	3.6E3	1.50E2	8.47E2
0900	1000	1.16E5	5.48E7	7.70E-4	4.22E4	3.6E3	1.52E2	9.99E2
1000	1100	1.14E5	5.36E7	7.60E-4	4.07E4	3.6E3	1.47E2	1.15E3
1100	1200	1.08E5	5.11E7	7.46E-4	3.81E4	3.6E3	1.37E2	1.28E3
1200	1300	1.08E5	5.11E7	7.40E-4	3.78E4	3.6E3	1.36E2	1.42E3
1300	1400	1.08E5	5.11E7	7.24E-4	3.70E4	3.6E3	1.33E2	1.55E3
1400	1500	1.08E5	5.11E7	6.98E-4	3.57E4	3.6E3	1.28E2	1.68E3
1500	1600	1.08E5	5.11E7	6.99E-4	3.67E4	3.6E3	1.29E2	1.81E3
1600	1700	1.08E5	5.11E7	6.80E-4	3.48E4	3.6E3	1.25E2	1.93E3

TABLE A-I (cont'd)

Date & Time From	To	Cubic Feet/ Minute	cc/second	µCi/cc	µCi/second	Sample Time/ Second	Total Curies	Curies (Running Total)
7/7 @ 1700	7/7 @ 1800	1.11E5	5.23E7	6.66E-4	3.49E4	3.6E3	1.26E2	2.06E3
1800	1900	1.11E5	5.23E7	6.63E-4	3.47E4	3.6E3	1.25E2	2.18E3
1900	2000	1.11E5	5.23E7	6.45E-4	3.38E4	3.6E3	1.22E2	2.31E3
2000	2100	1.11E5	5.23E7	5.88E-4	3.08E4	3.6E3	1.11E2	2.42E3
2100	2200	1.14E5	5.36E7	4.03E-4	2.16E4	3.6E3	7.78E1	2.50E3
2200	2300	1.14E5	5.36E7	3.39E-4	1.82E4	3.6E3	6.54E1	2.56E3
2300	2400	1.14E5	5.36E7	5.08E4	2.72E4	3 6E3	9.80E1	2.66E3
							Daily Total	2,660 Ci

TABLE A-1 (cont'd)

Date & Time From	Date & Time To	Cubic Feet/ Minute	cc/Second	μCi/cc	μCi/Second	Sample Time/ Second	Total Curies	Curies (Running Total)
7/8 @ 0000	7/8 @ 0100	1.16E5	5.48E7	4.70E-4	2.58E4	3.6E3	9.28E1	9.28E1
0100	0200	1.15E5	5.48E7	5.66E-4	3.10E4	3.6E3	1.12E2	2.05E2
0200	0300	1.15E5	5.48E7	5.80E-4	3.18E4	3.6E3	1.14E2	3.19E2
0300	0400	1.15E5	5.48E7	5.72E-4	3.14E4	3.6E3	1.13E2	4.32E2
0400	0500	1.15E5	5.48E7	5.58E-4	3.06E4	3.6E3	1.10E2	5.42E2
0500	0600	1.15E5	5.48E7	5.49E-4	3.01E4	3.6E3	1.08E2	6.50E2
0600	0700	1.15E5	5.48E7	2.13E-4	1.17E4	3.6E3	4.20E1	6.92E2
0700	0800	1.18E5	5.55E7	7.59E-6	4.21E2	3.6E3	1.51E0	6.94E2
0800	0900	1.18E5	5.55E7	6.01E-6	3.33E2	3.6E3	1.20E0	6.95E2
0900	1000	1.16E5	5.48E7	5.27E-6	2.89E2	3.6E3	1.04E0	6.96E2
1000	1100	1.16E5	5.48E7	4.91E-6	2.69E2	3.6E3	9.70E-1	6.97E2
1100	1200	1.14E5	5.36E7	4.56E-6	2.44E2	3.6E3	8.80E-1	6.98E2
1200	1300	1.15E5	5.42E7	1.07E-3	5.81E4	3.6E3	2.09E2	9.07E2
1300	1400	1.16E5	5.48E7	2.10E-3	1.15E5	3.6E3	4.15E2	1.32E3
1400	1500	1.14E5	5.36E7	1.85E-3	9.91E4	3.6E3	3.57E2	1.68E3
1500	1600	1.15E5	5.42E7	2.20E-3	1.19E5	3.6E3	4.30E2	2.11E3
1600	1700	1.15E5	5.42E7	2.40E-3	1.30E5	3.6E3	4.69E2	2.58E3

TABLE A-I (cont'd)

Date & Time From	To	Cubic Feet/ Minute	cc/second	μCi/cc	μCi/second	Sample Time/ Second	Total Curies	Curies (Running Total)
7/8 @ 1700	7/8 @ 1800	1.16E5	5.48E7	2.27E-3	1.24E-5	3.6E3	4.47E2	3.03E3
1800	1900	1.16E5	5.48E7	2.04E-3	1.12E5	3.6E3	4.02E2	3.43E3
1900	2000	1.16E5	5.48E7	1.96E-3	1.08E5	3.6E3	3.88E2	3.82E3
2000	2100	1.16E5	5.48E7	1.67E-3	9.18E4	3.6E3	3.30E3	4.15E3
2100	2200	1.19E5	5.61E7	1.46E-3	8.18E4	3.6E3	2.95E2	4.44E3
2200	2300	1.19E5	5.61E7	1.09E-3	6.13E4	3.6E3	2.21E2	4.66E3
2300	2400	1.19E5	5.61E7	1.18E-3	6.64E4	3.6E3	2.39E2	4,90E3
							Daily Total	4,900 Ci

TABLE A-I (cont'd)

Date & Time From	Date & Time To	Cubic Feet/ Minute	cc/Second	μCi/cc	μCi/second	Sample Time/ Second	Total Curies	Curies (Running Total)
7/9 @ 0000	7/9 @ 0100	1.16E5	5.48E7	7.16E-4	3.92E4	3.6E3	1.41E2	1.41E2
0100	0200	1.14E5	5.38E7	1.60E-4	8.57E3	3.6E3	3.09E1	1.72E2
0200	0300	1.14E5	5.38E7	1.60E-4	8.57E3	3.6E3	3.09E1	2.03E2
0300	0400	1.14E5	5.38E7	1.55E-4	8.30E3	3.6E3	2.99E1	2.33E2
0400	0500	1.14E5	5.38E7	3.34E-4	1.79E4	3.6E3	6.44E1	2.97E2
0500	0600	1.19E5	5.61E7	1.02E-3	5.71E4	3.6E3	2.06E2	5.03E2
0600	0700	1.16E5	5.48E7	3.60E-4	1.98E4	3.6E3	7.11E1	5.74E2
0700	0800	1.19E5	5.61E7	7.46E-4	4.18E4	3.6E3	1.51E2	7.25E3
0800	0900	1.18E5	5.55E7	7.63E-4	4.23E4	3.6E3	1.52E2	8.77E2
0900	1000	1.16E5	5.48E7	7.52E-4	4.12E4	3.6E3	1.48E2	1.03E3
1000	1100	1.16E5	5.48E7	7.16E-4	3.92E4	3.6E3	1.41E2	1.17E3
1100	1200	1.16E5	5.48E7	5.91E-4	3.24E4	3.6E3	1.17E2	1.28E3
1200	1300	1.11E5	5.23E7	4.57E-4	2.39E4	3.6E3	8.62E1	1.37E3
1300	1400	1.11E5	5.23E7	4.28E-4	2.24E4	3.6E3	8.07E1	1.45E3
1400	1500	1.08E5	5.11E7	3.43E-4	1.75E4	3.6E3	6.30E1	1.51E3
1500	1600	1.10E5	5.17E7	2.63E-4	1.36E4	3.6E3	4.90E1	1.56E3
1600	1700	1.11E5	5.23E7	2.34E-4	1.23E4	3.6E3	4.42E1	1.61E3

TABLE A-1 (cont'd)

Date & Time From	Date & Time To	Cubic Feet/ Minute	cc/second	µCi/cc	µCi/second	Sample Time/ Second	Total Curies	Curies (Running Total)
7/9 @ 1700	7/9 @ 1800	1.11E5	5.23E7	2.05E-4	1.07E4	3.6E3	3.87E1	1.65E3
1800	1900	1.15E5	5.42E7	2.91E-4	1.58E4	3.6E3	5.68E1	1.70E3
1900	2000	1.19E5	5.61E7	3.54E-4	1.99E4	3.6E3	7.15E1	1.77E3
2000	2100	1.19E5	5.61E7	2.15E-4	1.21E4	3.6E3	4.35E1	1.82E3
2100	2200	1.19E5	5.61E7	1.32E-4	7.42E3	3.6E3	2.67E1	1.84E3
2200	2300	1.21E5	5.73E7	8.23E-5	4.72E3	3.6E3	1.70E1	1.86E3
2300	2400	1.21E5	5.73E7	5.22E-5	2.99E3	3.6E3	1.08E1	1.87E3
							Daily Total	1,870 Ci

TABLE A-1 (cont'd)

Date & Time From	Date & Time To	Cubic Feet/ Minute	cc/Second	[illegible]	µCi/Second	Sample Time/ Second	Total Curies	Curies (Running Total)
7/10 @ 0000	7/10 @ 0100	1.[illegible]4E5	5.86E7	3.35E-4	1.98E3	3.6E3	7.13E0	7.13E0
0100	0200	1.19E5	5.61E7	2.00E-[illegible]	1.12E3	3.6E3	4.03E0	1.12E1
0200	0300	1.14E5	5.36E7	3.82E-6	2.05E2	3.6E3	7.37E-1	1.19E1
0300	0400	1.14E5	5.36E7	3.49E-6	1.87E2	3.6E3	6.73E-1	1.26E1
0400	0500	1.14E5	5.36E7	3.[illegible]E-6	1.75E2	3.6E3	6.30E-1	1.32E1
0500	0600	1.14E5	5.36E7	3.05E-6	1.63E2	3.6E3	5.88E-1	1.38E1
0600	0700	1.14E5	5.36E7	2.86E-6	1.53E2	3.6E3	5.52E-1	1.43E1
0700	0800	1.14E5	5.36E7	2.61E-6	1.40E2	3.6E3	5.03E-1	1.49E1
0800	0900	1.17E5	5.30E7	[illegible]76E-6	1.46E2	3.6E3	5.26E-1	1.54E1
0900	1000	1.11E5	5.[illegible]3E7	2.57E-6	1.35E2	3.6E3	4.84E-1	1.59E1
1000	1100	1.11E5	5.23E7	[illegible]	1.39E2	3.6E3	4.99E-1	1.64E1
1100	1200	1.11E5	5.23E7	[illegible]	1.39E2	3.6E3	5.01E-1	1.69E1
1200	1300	1.06E5	4.99E7	[illegible]	1.51E2	3.6E3	5.43E-1	1.74E1
1300	1400	1.06E5	4.99E7	[illegible]	1.63E2	3.6E3	5.86E-1	1.80E1
1400	1500	1.06E5	4.99E7	[illegible]	1.79E2	3.6E3	6.45E-1	1.86E1
1500	1600	1.14E5	5.36E7	[illegible]	8.68E2	3.6E3	3.12E0	2.18E1
1600	1700	1.19E5	5.61E7	[illegible]	1.49E3	3.6E3	5.38E0	2.71E1

TABLE A-1 (cont'd)

Date & Time From	Date & Time To	Cubic Feet/ Minute	cc/second	µCi/cc	µCi/second	Sample Time/ Second	Total Curies	Curies (Running Total)
7/10 @ 1700	7/10 @ 1800	1.19E5	5.61E7	1.86E-5	1.04E3	3.6E3	3.75E0	3.09E1
1800	1900	1.19E5	5.61E7	1.37E-5	7.67E2	3.6E3	2.76E0	3.37E1
1900	2000	1.21E5	5.73E7	1.03E-5	5.89E2	3.6E3	2.12E0	3.58E1
2000	2100	1.19E5	5.61E7	8.27E-6	4.64E2	3.6E3	1.67E0	3.74E1
2100	2200	1.19E5	5.61E7	6.38E-6	3.58E2	3.6E3	1.29E0	3.87E1
2200	2300	1.21E5	5.73E7	6.17E-6	3.54E2	3.6E3	1.27E0	4.00E1
2300	2400	1.24E5	5.86E7	5.63E-6	3.30E2	3.6E3	1.19E0	4.12E1
							Daily Total	41.2 Ci

TABLE A-1 (cont'd)

Date & Time From	Date & Time To	Cubic Feet/Minute	[illegible]	[illegible]	[illegible]	Sample Time/Second	Total Curies	Curies (Running Total)
7/11 @ 0000	7/11 @ 0100	1.24E5	5.86E7	5.09E-6	2.98E2	3.6E3	1.07E0	1.07E0
0100	0200	1.24E5	5.86E7	[illegible]	2.34E2	3.6E3	8.47E-1	1.92E0
0200	0300	1.24E5	5.86E7	4.37E-6	2.56E2	3.6E3	9.22E-1	2.84E0
0300	0400	1.24E5	5.86E7	3.89E-6	2.28E2	3.6E3	8.21E-1	3.66E0
0400	0500	1.24E5	5.86E7	3.93E-6	2.30E2	3.6E3	8.29E-1	4.49E0
0500	0600	1.24E5	5.86E7	[illegible]	2.21E2	3.6E3	7.97E-1	5.29E0
0600	0700	1.24E5	5.86E7	3.51E-6	2.06E2	3.6E3	7.41E-1	6.03E0
0700	0800	1.24E5	5.86E7	3.31E-6	1.94E2	3.6E3	6.99E-1	6.73E0
0800	0900	1.24E5	5.86E7	3.31E-6	1.94E2	3.6E3	6.99E-1	7.42E0
0900	1000	1.14E5	5.36E7	2.39E-6	1.28E2	3.6E3	4.62E-1	7.89E0
1000	1100	1.06E5	5.11E7	1.10E-6	5.60E1	3.6E3	2.02E-1	8.09E0
							Daily Total	8.09 Ci

TABLE A-2

DAILY TOTALS FOR Kr-85 PURGE FROM June 28, 1980 to July 11, 1980

DATE	DAILY CURIES	TOTAL CURIES
6/28	351	351
6/29	1,330	1,681
6/30	4,141	5,822
7/1	6,687	12,509
7/2	4,952	17,461
7/3	5,791	23,252
7/4	5,817	29,069
7/5	3,929	32,998
7/6	3,817	36,815
7/7	2,658	39,473
7/8	4,900	44,373
7/9	1,872	46,245
7/10	41	46,286
7/11	8	46,294