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TRANSMITTAL OF SIXTH TAAG REPORT - Hmb-13-84

Dear Sirs:

Enclosed herewith is the sixth report of the Technical Advisory and Assistance Group (TAAG). This report covers the TAAG activities during the period from April 1, 1983 to August 1, 1983.

Very truly yours,

H M Burton

H. M. Burton, Manager
Technical Information & Examination Program

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SIXTH REPORT

OF THE

Technical Advisory and Assistance Group (TAAG)

April 1, 1983 to August 1, 1983

TAAG MEMBERS

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Sixth Report of TAAG

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Sixth Report of TAAG

I. Introduction

Pursuant to a letter from B. K. Kanga, Director, TMI-2, to W. H. Hamilton, Chairman, TAAG, dated April 11, 1982, the following technical matters were addressed by TAAG during the period from April 1, 1983 to August 1, 1983:

1. Investigate methods to determine the state of the lower region of the core.

2. Continue assistance in leadscrew examinations and in interpretation of Quick Scan results.

3. Assist in formulating water clean-up and defueling plan and strategy.

4. Continue to provide assistance related to the TAAG recommendations regarding reactor building decontamination and characterization:

    a. Plans for cleaning 282' level.
    b. Preparation of Radiological Engineering Plans.
    c. Increase air flow and filtration.
    d. Use of tunnel concept.
    e. Appraisal of shielding effects from water in 282' level.

5. Assist in development of plenum removal plan; resolve items on pre-requisite list for plenum removal.

6. Appraise system for criticality control for reactor disassembly and defueling.

7. Examine design of and need for an on-line gamma monitoring system for the reactor building.

8. Review the EG&G core examination plan and matrix, regarding the utility of the plan in benefiting or impacting core defueling.

This report is organized to report progress on each of these items as a separate section of the report. Recommendations are included in each section of the report where the investigatory work has been completed.

II. Lower Region of the Core

TAAG evaluations of methods to determine the state of the lower region of the core covered (1) an assessment of ultrasonic and sonar techniques for such examination and (2) TV examinations of the lower core region. In summary:

A. Sonar Techniques

There has been some experience in using sonar pulsing techniques to determine the nature and extent of river sludge and other surfaces. With this experience in mind, the Electric Boat member of TAAG discussed the technique with several people and organizations experienced in the technique to determine if it would be useful in exploring the TMI-2 rubble bed. The scope and results of these efforts are discussed in Appendix II-1.

The conclusion of the effort is that use of the sonar technique in exploring the condition of the TMI-2 core would not produce substantive information.

B. TV Examinations of the Lower Core Region

A preliminary evaluation summarized in the Fifth TAAG report indicated that it should be possible to lower a TV camera down the annulus between the core barrel and the reactor vessel. This would permit visual inspection of the region of the lower reactor vessel head, as well as the inspection of various core support assembly bolted connections.

Funding to proceed with more detailed evaluations was received near the end of the current quarter, and the evaluations were initiated. Results of these evaluations will be discussed in the Seventh TAAG Report.

III. Lead Screw Examinations and Quick Scan Results

A. Lead Screw Examinations

A section of H-8 lead screw was sent by GPU to B&W Lynchburg Research Lab for investigation of any particulate and cesium deposits on the sample. The significant finding is that the cesium present on the sample is in a tightly adherent film on the metal, removable with nitric hydroflouric acid rinse but not with water. While further specimens will be examined, this result will mean that the underhead flushing program may only remove loosely adherent material but not the cesium bearing film.

The other section was sent to PNL for determining the presence of pyrophoric material. Initial results indicate that such material is not present.

Further evaluations from specimens from H-8 and other lead screws are being considered. TAAG recommends, however, that the section of lead screw support tube, which is removed during preparation for the Under-Head Exam program, be examined carefully.

B. Quick Scan Results

TAAG evaluations of the Quick Scan experiment performed in December, 1982 were included in the Fifth TAAG report. Assistance will be provided once the Quick Scan 2 experiment is performed. This experiment is currently scheduled to take place in September 1983.

As discussed in the Fifth TAAG Report (see Attachment VIII-TWO), one discrepancy between the TAAG evaluations and the GPUN evaluations of the Quick Scan I results is the predicted dose rate at the "planning basis" location. This location is at a distance four feet beyond the periphery of the reactor vessel inside diameter, and five feet above the vessel flange. The dose rate predicted by TAAG at this location once the reactor vessel head is removed is 40 r/hr, while the GPUN predicted dose rate is about 20 r/hr.

Some effort was spent during the current quarter in attempting to resolve this difference. The conclusions of that effort are:

- ° There is a difference of approximately two between the amounts of surface contamination estimated to be present on plenum components, based on the measured Quick Scan dose rates.

- ° This difference is due to a value for a constant employed in the evaluations called the "buildup factor", where TAAG and GPUN employed different data sources. This difference is not readily resolved.

- ° For conservatism, the high predicted dose rate should be employed where needed for planning purposes.

- ° The results of the Quick Scan II experiments should resolve any differences.

C. Under-Head Examination Program

The present GPU plan and procedure for mechanism removal calls for untorquing the 8 bolts which fasten to the mechanism to the closure head mechanism flange. Up to 2500 ft-lbs torque is permitted by the procedure to untorque the bolts. Since there has been a history of these bolts sticking and stripping threads on other plants, there is a liklihood some of the TMI-2 bolts will stick. But in the TMI case it is planned to re-use the female threaded piece for installing a cap on the flange. Hence, it is important not to strip the threads by over-torquing. Hence, it is recommended that untorquing be limited to use of 400-500 ft-lbs on the tools.

IV. Water Cleanup and Defueling

TAAG has reviewed the GPU/Bechtel approach to the Defueling Water Cleanup System presented in Planning Study TPO/TMI-046, Rev. 0, April 1983 and Technical Plan TPO/TMI-047, Rev. 0, April 1983. The GPU/Bechtel water cleanup system is designed to restore water clarity after 20 hours following a sudden release of 300 lbs. of fine (< 40 μm) particles. Also reviewed was an alternate proposal prepared for TAAG by Burns & Roe and presented in B&R Technical Memorandum TM3680-9, Rev. 1, June 20, 1983. Table IV-1 is a side by side comparison of the two approaches.

The differences between the two systems are the result of different approaches to the same problem. The GPU/Bechtel approach seeks to develop a full capability system independent of defueling constraints; whereas the B&R approach seeks to make maximum utilization of existing, or out of reactor building equipment and to develop a defueling strategy which fits this system. From these two approaches come the following areas of agreement:

- the need for filtration and ion exchange capabilities for the Reactor Vessel and for the fuel pools independent of the defueling system.
- the need for surge capability for feed to the ion-exchange system, which will remove spikes of dissolved radionuclides.
- the need for water clarity during mechanical defueling operations, i.e., all defueling operations other than hydraulic fuel removal.
- the need for a rapid (~ 24 hours) recovery from outbreaks of activity during defueling operations.
- the need for rapid (~ 24 hours) turn over of fuel pool water inventory to recover from outbreaks of activity during fuel handling operations.

TABLE IV-I

COMPARISON OF WATER CLEANUP SYSTEMS

ITEM	GPU/BECHTEL	B&R
RV CLEANUP SYS.		
CLEANUP VOLUME	78,000 GAL.	17,000 GAL.
FILTRATION RATE	400 GPM	150 GPM
ION EXCHANGE RATE	20-60 GPM	20-150 GPM
LOCATION	RB/FHB/AB	FHB/AB
CONSTRUCTION	ALL NEW SYSTEM NEW POWER & CONTROL MOST WORK IN RAD AREA	SLIGHT MODIFICATIONS EXISTING POWER & CONTROL LITTLE WORK IN RAD AREA
LEAD TIME	1 - 2 YEARS	2 - 3 MONTHS
DESIGN LIMITS	NOVEL FILTER DESIGN FUEL REMOVAL	LIMITED WATER VOL.
FUEL POOL/TRANSFER CANAL CLEANUP SYS.		
CLEANUP VOLUME	764,000 GAL.	400,000 GAL.
FILTRATION RATE	400 GPM	150/300 GPM
ION EXCHANGE RATE	20-60 GPM	20-150 GPM
LOCATION	FHB/AB	FHB/AB
CONSTRUCTION	MODS TO SF SYSTEM	UTILIZES RV CLEANUP SYS.
LEAD TIME	2 - 3 MONTHS	EXISTS
DESIGN LIMITS	DESIGN MODS TO FILTERS	USES RV CLEANUP EFFLUENT

Also, an area of lesser agreement is the acceptibility of "blind defueling" for hydraulic fuel removal effort. It is the position of several members of TAAG that such an approach will work and will permit expeditious removal of rubblized fuel debris. GPU/Bechtel considers water clarity a prerequisite for all defueling operations but would not require it for the hydraulic defueling operation if it proves to be an operational liability.

A serious concern TAAG has with the GPU/Bechtel system is the reliance on a sintered metal filter as the prime solids removal device. The concern is ~~that such a filter~~ design could lead to operational problems, should plugging occur before the vessel is loaded or should frequent back-flushing be required, which could seriously reduce the design flow rate. An on-going testing program including an in-reactor test after head lift would be the most definitive preoperational determination of the solution. Should the proposed filter design not prove acceptable, suggested alternate filter designs include deep bed type filters. One good possibility is zeolite beds as a combined filter/ion exchange column.

A problem with both approaches is the possibility that of floating debris might obscure the view of the rubble bed. The downward flow established by either system will not be adequate to remove lighter debris. It may become necessary to introduce a skimmer, or some other suction point high in the reactor vessel in order to remove material floating or suspended near the surface.

Also, since the GPU/Bechtel cleanup system filter is designed to handle the relatively small quantity of fuel that is less than 40 μm in size, it should be protected by a gravity fall-out tank to prevent inadvertent loading with larger size fuel particles.

There was concern by TAAG that the water cleanup system is a prerequisite for plenum removal and, as such, could place an R&D program on the critical path for plenum removal. Due to the complexity of the design effort for the water cleanup system, TAAG would recommend that the following steps be taken:

1) Sample and inspect the plenum debris to determine if it will become suspended during plenum removal.

2) If the debris is suspendable, take steps to clean the plenum or to fix the debris to the plenum.

3) A small disposable filtration system may be acceptable to monitor water clarity around the plenum for this evolution.

4) Steps should be taken to make the cleanup system pre-assembled on skids with no need for construction in the canal after plenum removal.

The TAAG conclusion is that the GPU/Bechtel approach is sound, although very conservative. The system design should proceed with the expectation that alternate filter designs can be substituted should operation with the Mott sintered metal filters identify some unexpected deficiency. The B&R proposal can be viewed as a low cost alternate approach which can be implemented expeditiously at any time a cleanup system is required prior to the development and testing of the GPU/Bechtel system, or as an alternative should the GPU/Bechtel system development falter. Modifications necessary to the existing systems to implement the B&R proposal are not deemed to be difficult.

V. Reactor Building Decontamination

A. Data Requirements for the Decontamination of the 282'6" Elevation

The 282'6" elevation of the TMI-2 reactor building is contaminated to such an extent that conventional decontamination techniques and procedures would be extremely costly in terms of exposures, of effort and of resources. Before such a campaign is mounted, both the decision for the timing of the effort and the data required for the effort must be obtained.

TAAG sees no compelling reason to divert resources away from defueling activities to decontaminate the reactor building basement. No major defueling activities are necessary in the basement and, with few exceptions, the dose rates on the 305' and the 347'6" elevations is believed to be due to contamination on those elevations.

The exceptions are near the large penetrations through the 305' El. floor which permit direct shine from the basement. There are two approaches to shielding this source: 1) reflood the basement with "clean" water, or 2) place temporary shielding over all major floor penetrations on elevation 305'.

Reflooding the basement has the potential advantage of decontaminating with minimum personnel exposures or effort by leading activity off of and out of structures into solution but at a high waste disposal and processing cost. The floor of the basement will be covered with several inches of water which will provide some decontamination. This water will be processed periodically. B&R Technical Memorandum TM 3680-9, Rev. 1 shows a method of introducing this reflood water into the reactor building as well as a method of maintaining that shield water at a low concentration without effecting other decontamination or defueling activities. However, the reflood of the basement has been specifically rejected by GPU/Bechtel in the GPU Nuclear memorandum from J. C. Devine, Jr. to B. K. Kanga, No. 4500-83-0296, dated June 9, 1983.

The second option, the selected shielding of floor penetrations, has been adopted and is essentially completed. Hence, the dose rate contribution from the 282'6" El. to the 305' and 347'6" Els. is believed to be essentially eliminated.

Airborne contamination generation from the basement is an issue separate from dose rates. This phenomenon does not impact dose rates on the operating elevations but does increase the airborne and smearable contamination on these elevations. This effect can be reduced or eliminated by isolating air flow from the basement to the upper elevations. Efforts along this line are under consideration by GPU/Bechtel and TAAG supports these efforts.

The only identifiable operation for defueling that requires access to the basement is the opening of isolation valve SF-V104, located in the Northeast quadrant of the reactor building basement approximately 13 feet above the floor. This operation is necessary to permit the use of existing systems to fill and to process the water in the fuel transfer canal to support defueling. Robotic approaches to this problem or alternative approaches to the canal fill and processing are more cost effective than decontamination to support a manned entry.

Hence, TAAG recommends that no decontamination of the 282'6" El. be undertaken until after defueling. However, data acquisition efforts should proceed.

Data requirements for the eventual decontamination of the 282'6" El. are listed below:

1) Location and distribution of contamination
2) Chemical and physical form of contamination
3) Decontamination techniques
4) Structural limitations on destructive decontamination techniques
5) End point dose rate.

Determining the location and the distribution of contamination is necessary for the planning of any decontamination effort. Experience on elevations 305' and 347'6" indicates that simple decontamination procedures, such as hydrolasing, will not be effective on concrete surfaces. In order to develop effective techniques, data must be obtained to fully describe the contamination in the basement. Radiation surveys to date have been performed primarily with TLD strings lowered down from El. 305'. While informative, such data is dominated by high area dose rates and is open to interpretation. Directional surveys must be taken with columnated detectors to specifically identify the large contributors to the dose rates. This will be especially important on El. 282'6" because, for the first time in the reactor building entry program, piping, valves and components containing significant quantities of contamination will be encountered. Even normally non-radioactive components such as motors, and cabinents will be significant sources if they were flooded during the accident. Many of these will not respond to externally applied decontamination techniques and will need to be identified and shielded.

The major source in the basement is likely to be the concrete structures. The amount and distribution of that contamination need to be determined for each type of concrete in the building; hollow blocks, solid blocks, 3000 psi poured concrete, and 5000 psi poured concrete. The effect of paint on this internal contamination will need to be evaluated. As a first step, core bores of the fill slab (painted, 3000 psi concrete), the D-rings (painted 5000 psi concrete) and the impingement walls (painted and unpainted 5000 psi concrete) should be taken and examined. Other core bores should be taken as needed.

The bottom of the 305' elevation concrete floor, i.e., the ceiling of the 282'6" elevation should also be surveyed and sampled. Experience on 305' elevation suggests that a significant source may exist in the overhead. This source should be quantified by core borings from the 305' El. care must be taken with these core

samples to preserve the data. The diameter of the core should be from 1 1/2" to 2" to preserve the surface finish of the sample. Water cooled coring equipment should not be used due to the leachability of cesium out of concrete.

Core samples should be analyzed to determine the depth and the concetration profile of the contamination within them. Spectrographic analysis should be used to determine the isotopic makeup of the contamination. Chemical analyses should be performed to identify the chemical makeup of the contamination. Due to the solubility of cesium, leach rate tests should be performed to determine if the dose rates from concrete can be significantly reduced by keeping the surfaces wet for long periods of time. Leach rate testing must be performed through the exposed surface, painted or unpainted, in order to be representative for use on the 282'6" El.

Once the core samples and dose rate data are known, decontamination techniques must be evaluated both for effectiveness and for compatibility with liquid waste treatment systems available at the TMI-2. Destructive techniques, if required, must evaluate the impact on the integrity of the affected structures and must limit amounts of material removal.

Other sources identified by radiation surveys must be physically sampled to determine the makeup of the contamination. For items having commonality with items on the operating elevations, such as cable, cable trays, the liner, and the open stairway, data from the 305' El. should be adequate to decide which decontamination technique must be utilized. If the source is specific to the basement, samples should be taken to determine the best technique for decontaminating each source.

B. <u>Radiological Engineering Plans</u>

Comments on this matter were discussed in the Fifth TAAG Report of April 1, 1983. Further effort on this matter was not made during the current report period.

C. Air Flow and Filtration

A long standing recommendation of TAAG has been to increase the air flow and filtration in the reactor building in order to reduce airborne contamination. The Fifth TAAG report recommended that the second train of the reactor building purge and purification system be turned on in the purification mode and that suitable baseline data be taken to identify and to quantify the results. This was done in early June 1983 and resulted in an order of magnitude decrease in the airborne contamination as measured by continuous air monitor.

An order of magnitude reduction in the airborne contamination could not have resulted solely from the increase flow rate. Doubling the air flow and filtration rate should result in a factor of two reduction in the airborne contamination. Other factors must have contributed to the initial decrease in concentration.

The most likely factor is the concurrent reduction in the airborne generation rate. The likely cause of this reduction is the reduction of the amount of air supplied to the D-rings by the reactor building air cooling units. The air supplied to the D-rings from the cooling units exhausts in the lower elevations of the D-rings. If this air flow rate is greater than the purge flow rate, which is taken from the "B" D-ring, contaminated air will be forced out of the D-rings on the 347'6" elevation and will contribute to the general airborne contamination problem. If air can be prevented from leaving the D-rings to the 347'6" elevation, the general contamination rates could be reduced significantly.

Currently, efforts are on-going to isolate all forced air flow into the D-rings which will prevent air being forced out of the D-rings. TAAG supports this effort. Once air supply dampers to the D-rings are closed, the reactor building air cooling unit fans should be put back on line to increase air mixing inside the building to enhance the effectiveness of the purge system.

It remains to be demonstrated that efforts to minimize airborne contamination by realignment of the air handling systems will result in the elimination of the need for respirators. Currently, BZA airborne concentrations are approximately 40 times greater than the constant airborne concentrations and have not been affected by the air handling system realignment. Eliminating the redeposition rate by isolating air flow into the basement may eventually result in lower BZA concentrations. TAAG supports efforts in this direction.

D. Pathways Approach to Containment Areas and Work Enclosures

TAAG has proposed that a pathways approach to containment areas and work enclosures be considered. This work has been done comparing the advantages of the pathways/work enclosures concept, considering the work effort and radiation exposure required for setting up and dismantling the pathways/work enclosures. TAAG has reviewed GPU/Bechtel comments on the pathways/work enclosures in GPU Nuclear letter serial 4300-83/U-324 dated June 3, 1983, which were subsequently reviewed in a meeting with GPU/Bechtel on June 15, 1983. Agreement could not be reached at the meeting on the use of the pathways/work enclosures approach.

The overriding differences between GPU/Bechtel and TAAG seems to be that the pathways/work enclosures may have serious effects on the work in containment outside the pathways/work enclosures - which are mainly defueling support, decontamination, dose reduction, plant maintenance and plant surveillance activities.

There was an important area of agreement. TAAG had suggested air conditioning of the enclosures for reduction of heat stress to personnel working in the pathways/work enclosures. GPU/Bechtel has suggested that chillers be installed outside of the Reactor Building to chill the water to the Reactor Building Air Coolers which in turn would cool the entire building. TAAG is in complete agreement on this point. Chillers are conveniently available and can be installed in a non-radioactive environment for the overall benefit of the TMI-2 recovery program. However, such chillers have

a long lead time of about 1-1 1/2 years. These chillers must be procured on an expedited basis in order to be available for next summer.

With regard to the pathways/work enclosures approval, TAAG notes that gross overall decontamination efforts to date have not reduced the airborne contamination, surface radioactive contamination, or general area radiation dose rates to the point that work can be performed inside containment without the full extent of health physics control, including full anti-contamination clothing and respirators. This drastically reduces the rate of progress toward the main objective of defueling the reactor plant, and degrades the efficiency of personnel working in containment by several orders of magnitude. The pathways/work enclosure concept does include lead curtains along the pathways/work enclosure to reduce the general area radiation dose rates.

It is the TAAG position that decontamination efforts may never be effective enough to improve this situation substantially. Furthermore, it is the TAAG position that the pathways/work enclosures approach is practical and will substantially improve this situation for critical path work. It remains the TAAG position that the pathways/work enclosures should be installed now and work should proceed based on use of this concept. If gross decontamination efforts inside containment should eventually prove so successful that the pathways/work enclosures are no longer needed, they can be removed. The relatively small costs and radiation exposure would be offset by expedited work on the critical path of defueling the plant.

With regard to ALARA considerations, TAAG points out that the radiation exposure involved in decontaminating the containment building was not included in the determinations of personnel exposure. It is therefore questionable that rejection of the pathways/work enclosures approach is ALARA at present because the very feasibility of decontaminating the building is in question. Furthermore, the radiation exposure in setting up the pathways/work enclosures is offset to a substantial degree by improved efficiency of personnel working within the pathways/work enclosures.

The pathways/work enclosures approach has been proven in shipyard practice and will work. The essential point is that it will decouple decontamination of the containment building from the critical path work of defueling the plant.

VI. Plenum Removal

Prerequisites for Plenum Removal

Section IV of the fifth TAAG Report provided a suggested listing of prerequisites which are considered unique to making preparations for successful plenum removal. As a follow up to this submittal, conversations between TAAG members and the Task Leader for Reactor Disassembly and Defueling indicates that the recommendations are being adequately factored into the reactor disassembly and defueling plan.

Review of Plenum Assembly Removal (PAR) Preliminary Engineering

TAAG has continued to review the preliminary engineering for plenum assembly removal. A representative of TAAG was in attendance at a design review meeting which was held April 12, 1983 (Bechtel Conference Notes No. 190 documented proceedings) and provided comments relative to the preliminary engineering completed to date and the committed system design concept for the PAR system.

TAAG observations regarding preliminary engineering preparations for plenum assembly removal are as follows:

1. There is a significant difference between the design weight of the plenum and the expected actual weight. The actual weight was unknown at the time of the conference. Since the actual plenum weight will make a difference in the interpretation of the plenum lifting data obtained during head removal, as well as affect the interpretation of load cell readings during initial jacking, it is vital that the actual weight of the plenum be more accurately determined. Design Engineering agreed to initiate a search for the site operational records which may provide the actual plenum assembly weight, as weighed by the TMI-2 polar crane during installation of the plenum during plant construction.

2. The Essential and Non-Essential monitoring requirements should be reevaluated to consider plenum assembly failures that could make plenum removal more difficult. This is of particular importance during the initial 2-1/2" jacking operation. Continuous video and load cell monitoring should be employed during jacking operations with procedural requirements restricting continued operations if certain pre-determined acceptable weight and motion changes do not occur/exist as jacking progresses.

VII. Criticality Control

TAAG was requested to evaluate the various methods proposed to assure that criticality did not occur during the head, plenum, and core fuel removal operations. These methods all used soluble poison as means to assure shutdown. They differed in the method to be used to monitor the shutdown. The purpose of the TAAG review was to evaluate the advantages of each system and recommend the system best suited to the TMI core 2 evolutions.

During this reporting period a series of meetings were held. At these meetings, GPUN personnel addressed the subject of criticality control and monitoring of core shutdown. As a result of these meetings GPUN concluded that reactor shutdown monitoring would be achieved by monitoring the boron concentration in the reactor pressure vessel. Members of the Safety Advisory Board and TAAG participated in the meeting where this discussion was reached [Dr. W. R. Stratton (SAB), Dr. N. Rasmussen (SAB) and R. S. Brodsky (TAAG)]. They concurred with this conclusion.

This GPU action preempted the need for a separate TAAG review of this issue. TAAG concurs with the results of the GPU actions.

VIII. Gamma Monitoring System

The purpose of the study of an On-line Gamma Monitoring System is to determine if such a system is needed in the TMI-2 Reactor Building. To determine need, a review of regulatory requirements, license requirements, and operational needs were examined.

Regulatory Requirements/Guidance

10CFR19.12 states, "All individuals working in or frequenting any portion of a Restricted Area shall be kept informed of the storage, transfer, or use of radioactive material or of radiation in such portions of the Restricted Area."

10CFR20.20(b) states, "Each licensee shall make or cause to be made such surveys as (1) may be necessary for the licensee to comply with the regulations in this part, or (2) are reasonable under the circumstances to evaluate the extent of radiation hazards that may be present."

10CFR50, Appendix A, Criterion 61 - Fuel Storage and Handling and Radioactivity Control - The applicability of this criterion is as it relates to occupational radiation protection aspects of fuel storage, handling, and radioactive waste.

> Criterion 13 - Instrumentation and Control - Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to ensure adequate safety.

Regulatory Guide 8.8, C.2.g - On-Line Radiation Monitoring Systems reduce exposure of personnel who would have to enter area for survey if system not provided and system provides timely information regarding changes in dose rate in an area.

NUREG 0800, U. S. Nuclear Regulatory Commission Standard Review Plan, Chapter 12.3-12.4 - Radiation Protection Design Features I.4 - Requires description of system, criteria for placement, and location of monitors.

License Requirements/Basis

FSAR, Chapter 12.1.4 states, "The fixed Radiation Monitoring System is designed to indicate and record radiation levels throughout the unit...." Section 12.1.4.1 states, "The detector functions, locations, ranges, and set points... were selected to monitor normal plant operations, and to monitor and provide additional alarms during and following abnormal operations or accidents up through and including a Maximum Hypothetical Accident."

Technical Specifications - The fixed Radiation Monitoring System is not covered by Tech. Specs. Tech. Specs. do require the conduct of the Radiation Protection Program in accordance with 10CFR20.

General Project Design Criteria 13587-2-L01-100 - The criteria is applicable to systems and facilities for TMI-2 recovery. Design is influenced by concern for public and occupational health and safety. Clarifications are provided for referenced Regulatory Guides. Regulatory Guide 8.8, paragraph C.2.g clarification states, "In addition, area radiation monitors will be provided in areas to which personnel normally have access and where there is a potential for personnel unknowingly receiving high levels of radiation exposure (e.g., in excess of 10CFR20 limits) in a short period of time because of system failure or improper personnel action."

Operational Requirements

An On-Line Gamma Monitoring System is one method to provide a real time assessment of radiological conditions. An On-Line System can provide continuous gamma radiation monitoring, which informs plant personnel immediately when predetermined exposure rates are exceeded in various areas of the Reactor Building so appropriate action can be taken. An On-Line Gamma Monitoring System can provide indication of changing plant

conditions, trend analyses, and detect areas of high exposure rate to prevent exceeding limits of 10CFR20. This capability is particularly beneficial for the evolutions in the Reactor Building. Head removal, internals removal, and ultimately core removal can cause quickly changing radiological conditions which need to be detected in a timely manner to prevent unnecessary and high radiation exposures.

Present System Description

The On-Line Gamma Monitoring System presently in TMI-2 Reactor Building consists of six (6) detectors. They are located throughout the building. At present, the system is inoperative with exception of HPR-213. TABLE 1 provides a description, location, and status of each monitor. The FSAR states the locations were chosen to monitor normal plant operations and provides additional alarms during and following abnormal operations or accidents.

TABLE 1

AREA MONITOR DESCRIPTION

Channel No.	Location	Status
HP-R-209	F. H. Bridge North	Disconnected
HP-R-210	F. H. Bridge South	Disconnected
HP-R-211	Personnel Access Hatch	Removed
HP-R-212	Equipment Hatch	Removed and Replaced
HP-R-213	Incore Instrm. Panel Area	Removed and Replaced
HP-R-214	Reactor Building Dome	Removed

Conclusion

Further information/evaluations regarding the need for an on-line gamma monitoring system will be developed during the next report period.

IX. Core Examination Program

TAAG was requested to review the Core Examination Program prepared by the Department of Energy in conjunction with its Technical Evaluation Groups. It was reviewed in a TAAG meeting. (See Appendix IX-1). It is the intent of TAAG to keep this program of acquiring data for future core designs and analyses in mind as various defueling efforts are planned.

X. Summary of Recommendations

Page No.	Recommendation
4	Analysis of leadscrew support tube.
5	Limit torque applied to mechanism bolts during Under-Head Exam.
9	Sample and inspect plenum debris to determine if suspendable. If suspendable, clean or fix debris. Use small disposable filter. Clean-up system preassembled on skids.
9	Consider use of mini-decay heat system if clean-up system becomes too costly.
11	Decontamination of 282'6" level deferred till after defueling.
11-13	Data from 282'6" level Directional surveys Core bores of fill slab, D-rings and impingement walls. Bottom of 305' floor - survey and core bores. Core bores analyzed for depth and concentration, chemical compounds and leachibility. Based on core bore samples determine decon technique.
14	Supports increased air flow efforts and decreasing flow through basement.
15	Use of Pathways concept.
18	Use actual not design, weight of plenum.
19	Continuous video and load cell monitoring during plenum lift.

APPENDICES

APPENDIX II-1

GENERAL DYNAMICS

Electric Boat Division

Eastern Point Road, P.O. Box 1147
Groton, Connecticut 06340
Reactor Plant Services • 203 446-4300/4400

File No.: RPS-12242 June 8, 1983

Subject: Assessment of Ultrasonic and Sonar Techniques for TMI-II Core Examination

Enclosures:
(1) TAAG Presentation of 4/6/83
(2) Telecon Report of 5/13/83 between J. Brown/R. Grills and R. Muenow
(3) Characterization of TMI Reactor Core Material using Acoustic Methods

Mr. W. H. Hamilton
Post Office Box 613
Ligonier, PA 15658

Dear Mr. Hamilton:

General Dynamics/Electric Boat Division (GD/EBDiv) was requested by TAAG to examine the possibility of using either Ultrasonic or Sonar techniques for examining the damaged TMI-II core in an effort to make some projection of damaged core makeup to aid in tooling design. Experts in ultrasonics and sonar from the U.S. Naval Underwater Sound Laboratory, Raytheon, University of Rhode Island Ocean Engineering, Muenow and Associates and EPRI were also tasked with to verify the conclusions that were reached by General Dynamics/Electric Boat Division personnel.

A preliminary report (enclosure (1)) was made to TAAG on 4/6/83 regarding the feasibility of using both ultrasonics and sonar for probing the damaged core in order to help ascertain core structure with the intent of aiding defueling tooling design. The preliminary assessment was that the relatively high frequencies associated with ultrasonics did not have the penetrating powers to go through a potentially fractured mass but that parametric sonar techniques held some promise. At that meeting, use of UT for finding voids in concrete was discussed but further investigation (enclosure (2)) indicated that its applicability to the TMI core was negligible.

With task funding provided through EG&G, additional investigation was undertaken to further define the feasibility of sonic probing of the core. The results of this investigation are reported in enclosure (3). In summary however, it has been determined that the parametric sonar scanning technique, although holding some promise, is highly dependent on reasonably accurate modeling of the core in order to obtain meaningful engineering data. The wide range of potential core conditions, particularly in the region below the rubble bed, could lead to a wide range of data interpretation and consequently information of questionable value. Of particular concern are aggregate material density, entrapped gas pockets, and "layering" which, dependent on assumptions, could lead to widely varying interpretation of results.This, coupled with cost estimates ranging from $50,000 for a very simple proof of principle test to over $200,000 for a test which more accurately reflects reactor configuration, leads us to the conclusion that it should not be pursued further.

It is noted that virtually all the people we discussed this with felt that the only way to get the type of definitive information we were seeking on material properties and current core configuration was to physically penetrate the core. Electric Boat Division's opinion is that a core base (or drill with material collection coupled with boroscope examination) would be the least expensive and most definitive way to obtain the desired information. We see no reason why this could not be accomplished with the reactor head in place which allows it to be done fairly quickly so that the information obtained could be used in the early stages of tooling design.

Very truly yours,

GENERAL DYNAMICS
Electric Boat Division

TSCramer

T. S. Cramer
688 Program Manager

TAP/pb/0052S

cc: TAAG Members
H. Burton (EG&G)

Enclosure (1)

SONIC PROBING OF TMI-II CORE

I. INTRODUCTION

A. PURPOSE

B. MODEL USED

II. ULTRASONIC PROBING OF CORE

A. SYSTEM CONSIDERED

B. CONCLUSIONS

III. SONAR PROBING OF CORE

A. SYSTEM CONSIDERED

B. EXPECTED RESULTS

C. PROJECT SCOPE

I. INTRODUCTION

A. PURPOSE - TO INVESTIGATE VARIOUS SONIC METHODS FOR PROBING THE TMI-II CORE.

- DETERMINE THICKNESS AND DENSITY OF VARIOUS LAYERS.

B. MODEL USED

- o BASED UPON "QUICK LOOK" RESULTS
  - 5 FT. OF WATER BELOW PLENUM
  - 14" OF "LOOSE" MATERIAL
- o 3700 PPM BORON IN WATER
- o CHARACTERIZATION OF SOLID
  - o HOMOGENEOUS VS. NON HOMOGENEOUS CONCRETE LIKE TO HOMOGENEOUS CERAMIC/METAL
  - o DENSITY RANGE - 3 - 11 GM/CC

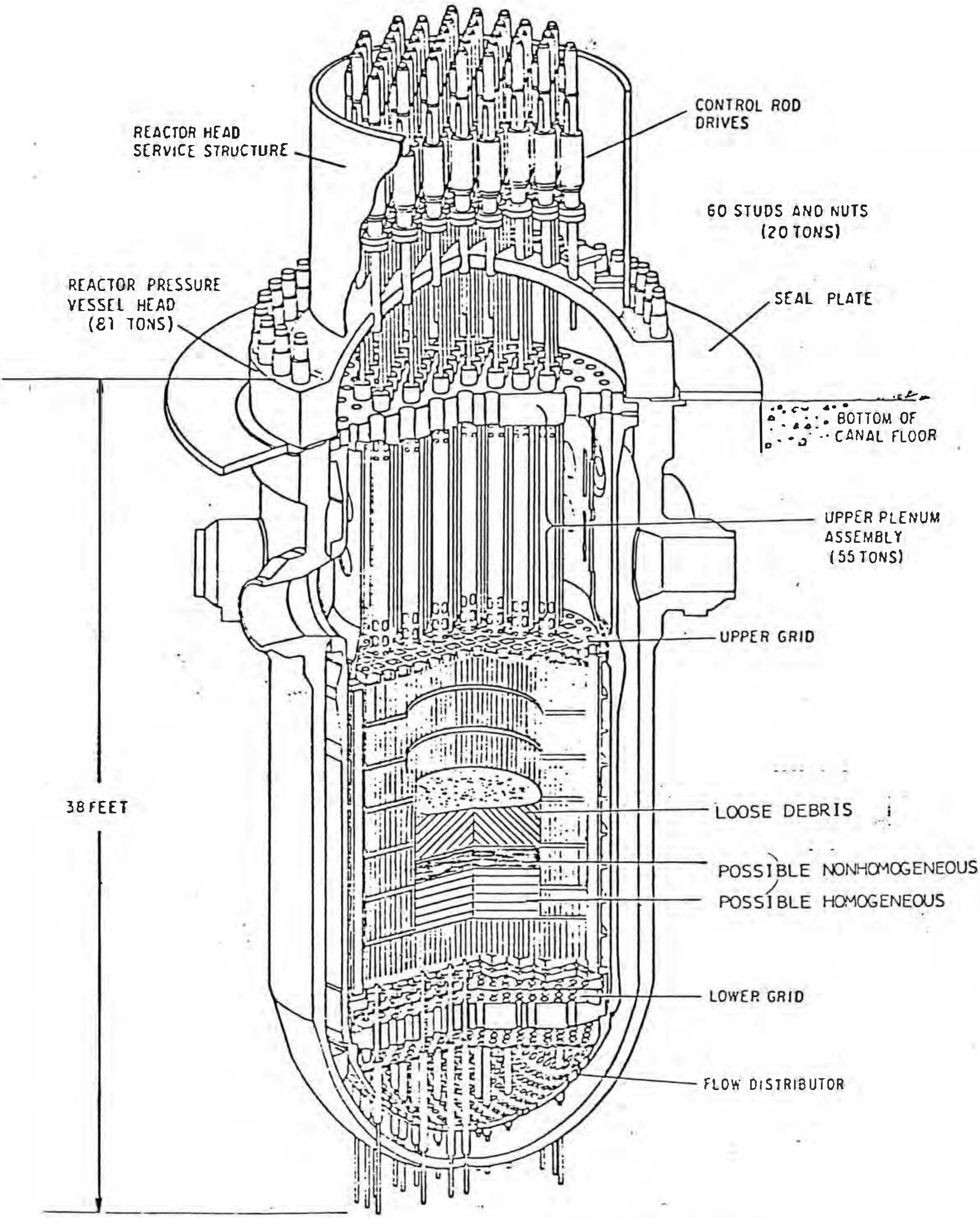


Figure 6.1. Cutaway View of a Typical PWR Similar to the TMI-2 Reactor Pressure Vessel.

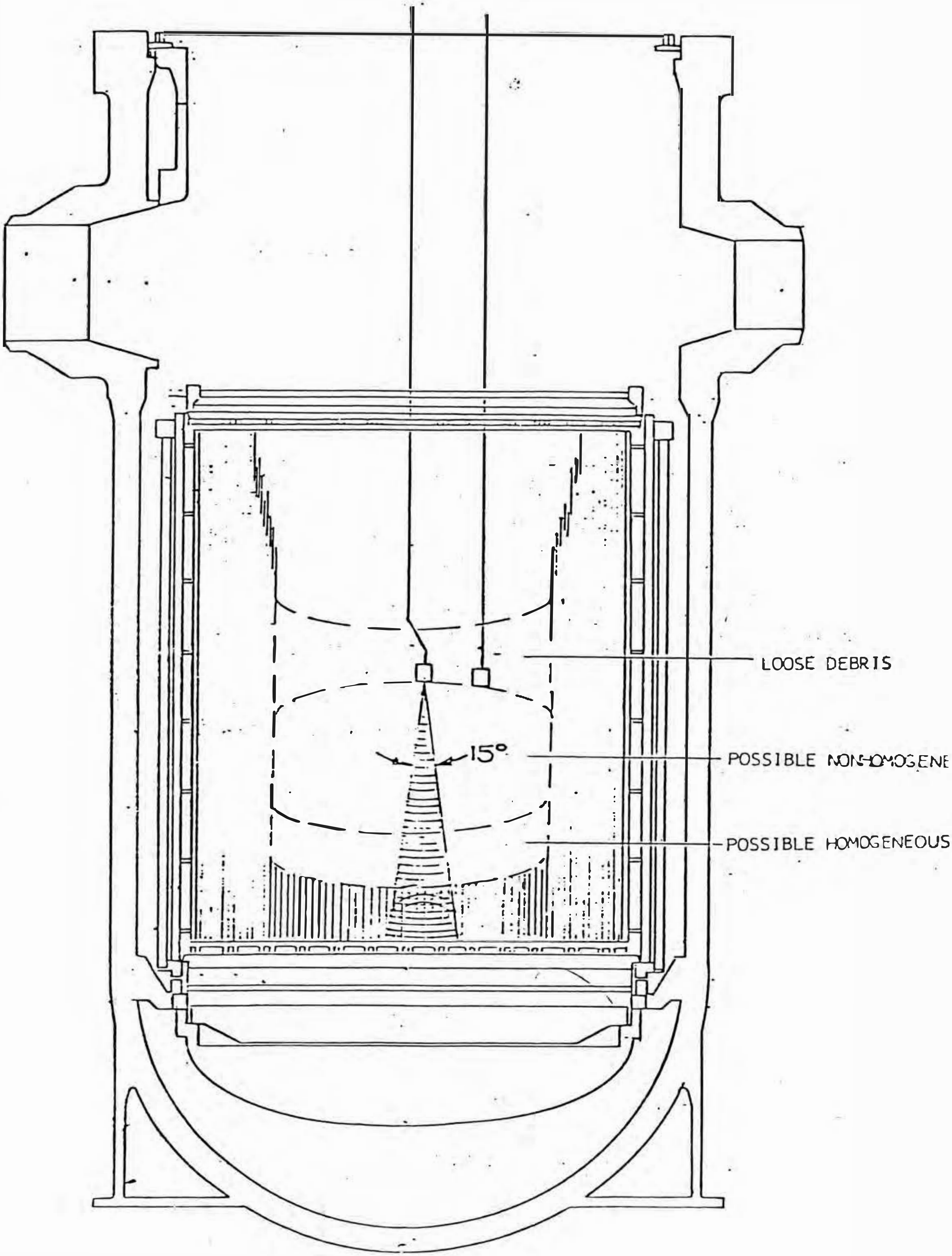
II. ULTRASONIC PROBING OF CORE

A. SYSTEM CONSIDERED

- o SIMPLE THICKNESS MEASURING SYSTEM
- o 200K Hz TO 500K
- o 1" DIA. MULTIPLE TRANSDUCERS
- o SYSTEM COST - LESS THAN $10,000

B. CONCLUSIONS

- o LOW DENSITY, NON-HOMOGENEOUS MATERIAL WILL ATTENUATE/SCATTER SIGNAL.
- o RELATIVELY LOW COST
  - - $10,000 FOR EQUIPMENT
  - - 3-4 MAN DAYS FOR PROOF-OF-PRINCIPLE
- o LESS THAN 25% CHANCE OF SUCCESS
  - - EBDIV. & EPRI CONCUR


LOOSE DEBRIS
15°
POSSIBLE NON-HOMOGENE
POSSIBLE HOMOGENEOUS

III. SONAR PROBING OF CORE

A. SYSTEM CONSIDERED - PARAMETRIC TRANSDUCER ARRAY

- 5K Hz TO 25K Hz DIFFERENCE FREQUENCY
- WATTS EFFECTIVE POWER
- FAR FIELD CONFIGURATION

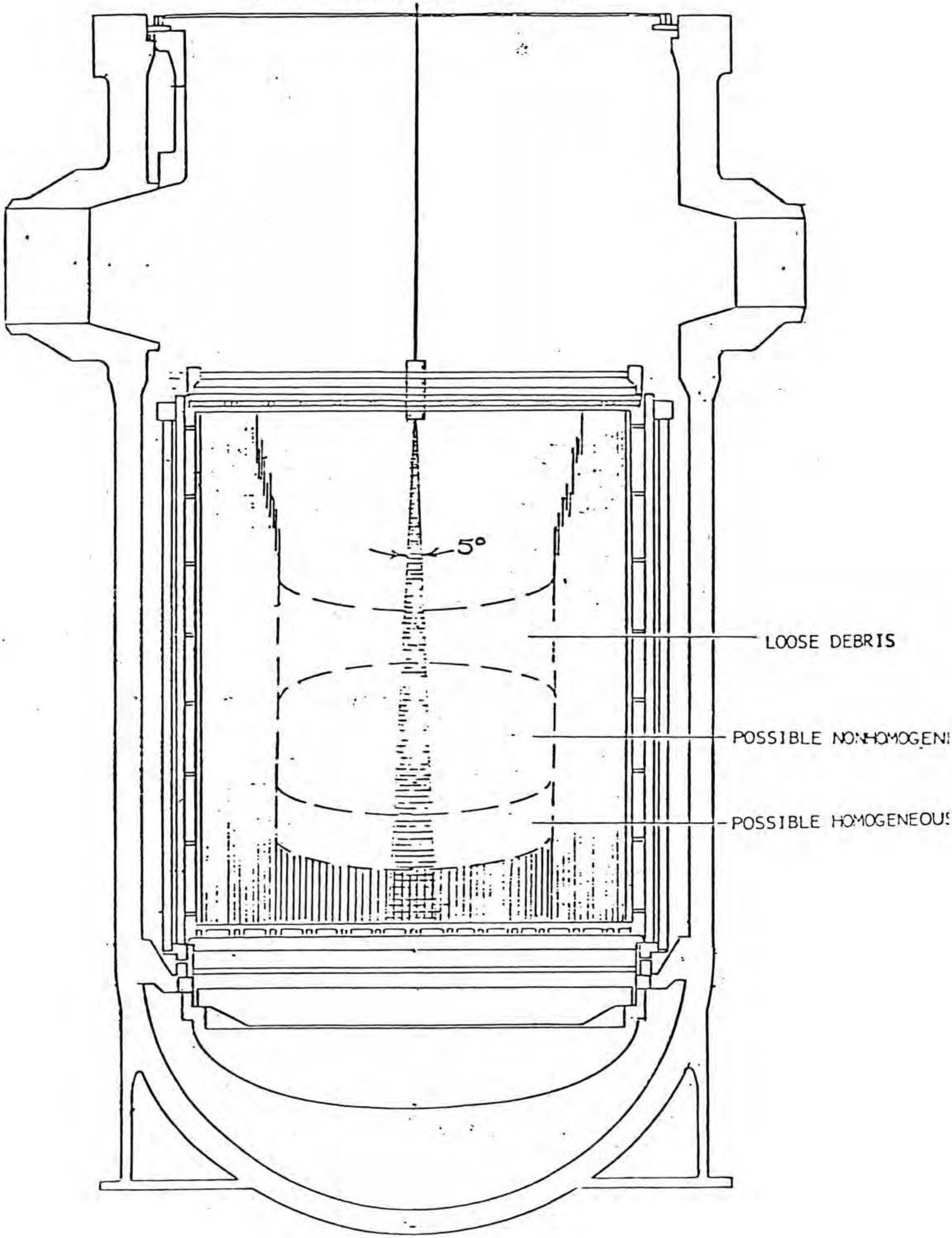
B. EXPECTED RESULTS

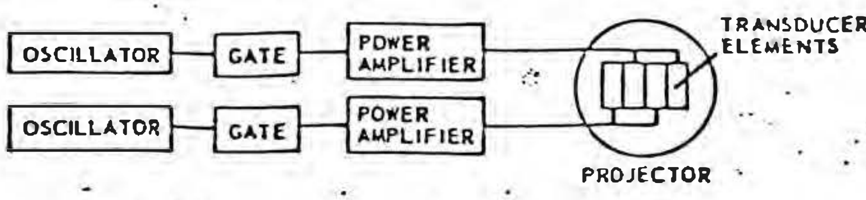
- WITHOUT FURTHER WORK. 60% SUCCESS PROBABLE
- THICKNESS RESOLUTION 1-2 INCHES DEPENDING ON DENSITY ACCURACY

C. PROJECT SCOPE

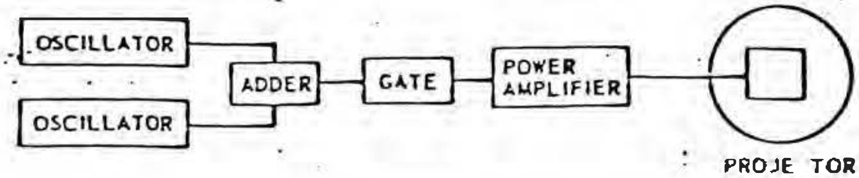
- PRELIMINARY ENGINEERING
- PROOF OF PRINCIPLE TEST
- ACTUAL

PARAMETRIC SONAR PROBING

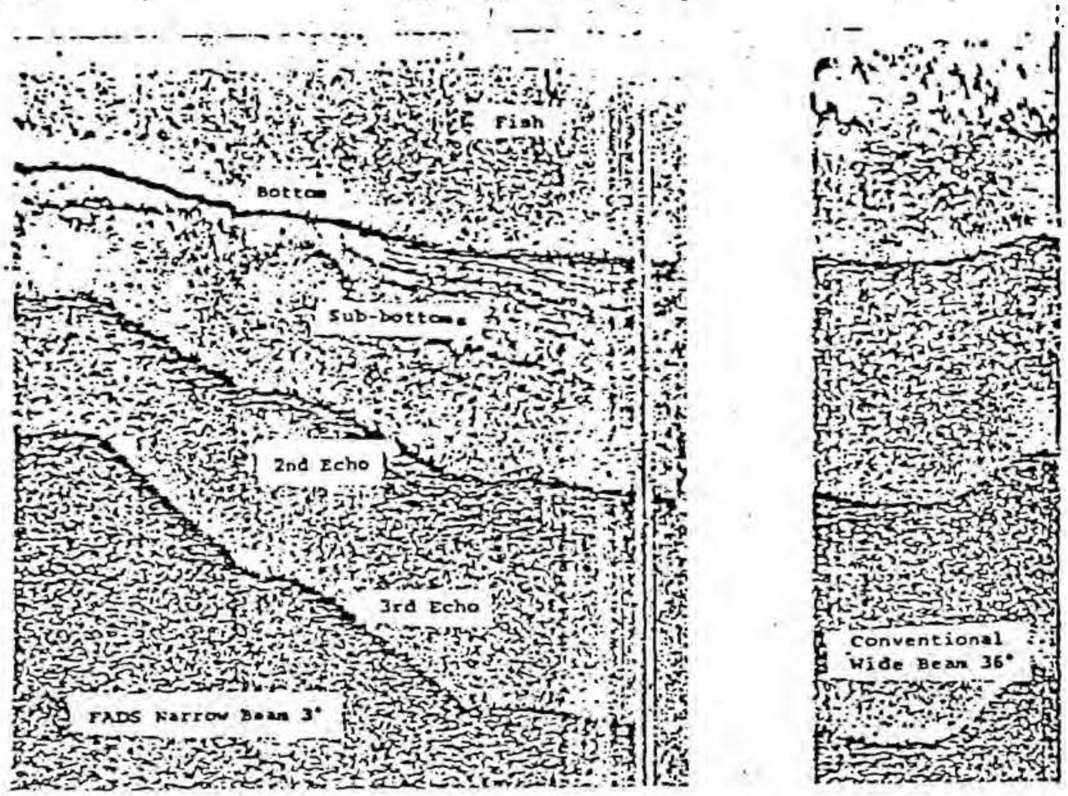




(A) DISCRETE PARAMETRIC TRANSMITTER



(B) COMPOSITE PARAMETRIC TRANSMITTER



TAKEN FROM: G WALSH, PROC. SPECIALISTS MEETING ON NONLINEAR ACOUSTICS, UNIV. BIRMINGHAM, APRIL 1971. BRITISH ACOUSTICAL SOCIETY.

FADS/CONVENTIONAL COMPARISON AT 12 kHz

GENERAL DYNAMICS
Electric Boat Division

TELECON/CONFERENCE REPORT

☐ CONFERENCE ☐ TELECON

FILE | PAGE 1

DATE & TIME OF MTG/TELECON: 5/13/83

PERSON INITIATING DISCUSSION	ACTIVITY/DEPT/CODE	LOCATION/ADDRESS	TELEPHONE NO.
J. Brown	EBDiv., Dept. 443	Groton, CT	(203)446-4600
PERSON CALLED			
R. Grills	EBDiv., Dept. 732	Groton, CT	(203)446-2620
OTHER PARTICIPANTS			
Richard Muenow	Muenow & Associates	Charlotte, NC	(704)377-4041

FOR CONFERENCE, IDENTIFY LOCATION

IF MTG/TELECON WAS WITH GOV'T AGENCY WAS LOCAL REP. NOTIFIED?
☐ NO ☐ NOT APPLIC. ☐ YES → ☐ SUPSHIP ☐ NRRO ☐

NAME(S) OR REASON FOR A "NO"

SUBJECT MATTER (INCLUDE AGREEMENTS, DECISIONS, COMMITMENTS, DIRECTIVES, APPROVALS, FOLLOW-UP REQUIRED)

Ultrasonic Testing (UT) of Concrete; Specification ASTMC-597-71

BACKGROUND

Mr. T. Cramer, Dept. 688, requested that the subject UT test methods used for QC of concrete be examined for potential use at the Three Mile Island Power Plant. It was thought that this UT method may be used to gain information on the character of the reactor vessel contents.

TELECON

1. Mr. Grills discussed the above with Dr. Steven Serabian at the Lowell Technical Institute. Dr. Serabian has extensive experience with UT test methods.

   Given that the vessel contents are largely unknown with regard to stratification and density of materials, Dr. Serabian stated that there is a low probability that the low frequency UT (10 to 100 KHz) could be used successfully. Further, the ASTMC method is a pulse velocity measuring technique which requires access for sensors on two surfaces. It is sometimes difficult to obtain useful data on concrete where the material composition is generally known. Some states have attempted to use the UT for QC measurements of concrete roadbeds and have abandoned this approach.

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DIVISION APPROVAL	DATE	AGENCY, TITLE, SIGNATURE		☐ NOT REQ'D ☐ REQ'D BY	☐ PRELIM ☐ REV. ☐ FINAL	

DISTRIBUTION:

2. I also called Mr. Richard Muenow of Muenow & Associates, Charlotte, N. C. This firm does extensive work on QC inspection of reactor plant materials especially of concrete. Unlike the ASTMC pulse velocity measuring system, Muenow uses a pulse echo system operating on frequencies between 10 and 150 KHz. Accordingly, access to only one surface is required. Their resolution is typically on the order of a 6- to 8-inch sphere in a depth of 6 to 12 feet of concrete. They have not operated in an environment such as inside of a vessel where reverberation could be a complication and, therefore, are not certain of the potential problems from reverberation effects.

JBrown:als
4600

CHARACTERIZATION OF THREE MILE ISLAND REACTOR CORE MATERIAL USING ACOUSTIC METHODS

Acoustic methods of non-intrusive examination of the TMI reactor core have been considered. These separate approaches were evaluated: conventional Ultrasonic (UT), conventional SONAR and parametric SONAR. Of these, the parametric sonar approach is considered the most viable.

The parametric array utilizes the nonlinear nature of the working fluid (water) to generate two modulation or sideband frequency components from the sum and difference of two primary source frequencies. See figure 1 for an example of reported performance. The parametric array has two characteristics which are very useful. When utilizing the difference frequency sideband, the size of the parametric transducer compared to conventional transducers is greatly reduced. Also, the parametric transducer can be designed to be exceptionally directional. These characteristics allow development in a confined area and assist in reducing signal returns from unwanted directions.

Even with this high directivity, the nature and geometry of the reactor containment vessel is highly unfavorable to exact characterization. There follows a description of a few of the important factors.

CONTAINED GEOMETRY

The reactor core material resides in a regular cylindrical shell. This shell will potentially reflect much of the vibrational energy in the core marerial causing a large number of possible paths (reverberation) for returned vibration energy. See figure 2.

NON-PARALLEL CORE MATERIAL LAYERS

Seismic profiling of ocean bottom sediments takes advantage of a roughly parallel and horizontal layering geometry to help identify the imaged data. The lack of physical experience with reactor failure material behavior restricts use of this assumption in general. It is expected, because of the symmetry of the containment vessel, that the center of the region may well have parallel, horizontal layers.

UNCERTAIN MATERIAL STATE

The highly complex thermodynamic/fluid dynamic process which occurred during the reactor incident makes estimation of material configuration and mixture highly difficult. Configurations such as uranium oxide pellets in a alloy flow or irregular sized aggregates will have highly variable acoustic sound speed and attenuation.

TRADE-OFF BETWEEN MATERIAL PENETRATION AND RESOLUTION

The closest material/geometry model for the reactor material examination known is oceanographic sediment characterization. Using the large volume of experimental data published, estimates may be made of the usable frequency ranges for acoustic imaging. Figure 3 shows representative data for several types of oceanographic measurement plus some low frequency data collected on land. Keeping in mind that approximately seventeen feet of penetration is desired, the data in figure 3 suggests that frequencies in the 3 khz to 30 khz range should be used to insure sufficient signal return. Since the imaging process would utilize a pulsed method, the resolution possible is controlled by two factors.

Figure 4 depicts the concepts of single reflection resolution and spacial resolution of a pulsed sinosoidal acoustic signal for multiple objects. Single reflection resolution describes how well the distance to a target can be defined. Generally, because of pulse alteration during reflection and propagation, measurement accuracy is estimated at one-half wavelength. Spacial resolution is depicted in figure 4b for two pulses with different pulse lengths.
The accuracy of measurement of an impedence discontinuity is at best for a highly reflective target equal to one half the wavelength of the pulse modulated sine wave.

$$1/2 = \frac{1}{2}\frac{C}{f}$$

For Uranium Oxide the bulk compressive sound speed is 5700 meters per second

$$\text{Reflective Resolution} = \frac{5700}{2f}$$

Assuming 30 khz as the difference frequency chosen

$$\text{Reflective Resolution} = \frac{5700}{60X10^3} = .095 \text{ Meters } (.31 \text{ feet})$$

A second consideration is the pulse duration of the transmittal signal. A reasonable range for satisfactory parametric array behavior is 50-100 cycles of the primary source frequency.

For the longer incident pulse, pulse A, no separation between echos is possible. For pulse B, the pulse length is sufficiently short to allow two discrete returns.

If, as in figure 1, 436 khz is the mean source frequency and 50 cycles are gated, then the time period of pulse, T is:

$$T = 50 \times \frac{1}{f} = \frac{50}{436X10^3} = 1.15X10^{-4}\text{sec}$$

At the difference frequency of 30 khz

$$T(30K) = \frac{1}{f} = \frac{1}{30X10^3} = 3.33X10^{-5}\text{sec}$$

The number of cycles of the 30 khz difference frequency is approximately

$$\#\text{ of Cycles} = \frac{T(\text{pulse})}{T(30K)} = \frac{1.15X10^{-4}}{3.33X10^{-5}} = 3.45$$

To develop sufficient pulse packet energy, roughly twice this number of cycles would be desirable; about

T pulse = $2.3X10^{-4}$ seconds

Length of pulse = 5700 m/sec x $2.3X10^{-4}$sec
(in UO_2) = 1.311 meters

This means that spacial resolution of 1.311 meters is what is practical using these numbers. At best assuming sufficient signal strength can be developed using only 3.45 cycles.

Spacial Resolution = .66 meters
in UO_2 (2.15 feet)

These simple calculations serve to demonstrate that trade-offs in penetration and resolution are necessary to obtain useful information. The spacial resolution and single surface reflection resolution are insufficient for this task. Higher difference frequencies and lower penetration would be required. Table 1 shows the material parameters and calculated bulk compressional and shear sound velocities of reactor materials. Information to calculate these values was provided by Dave Strauson of MPR Associates.

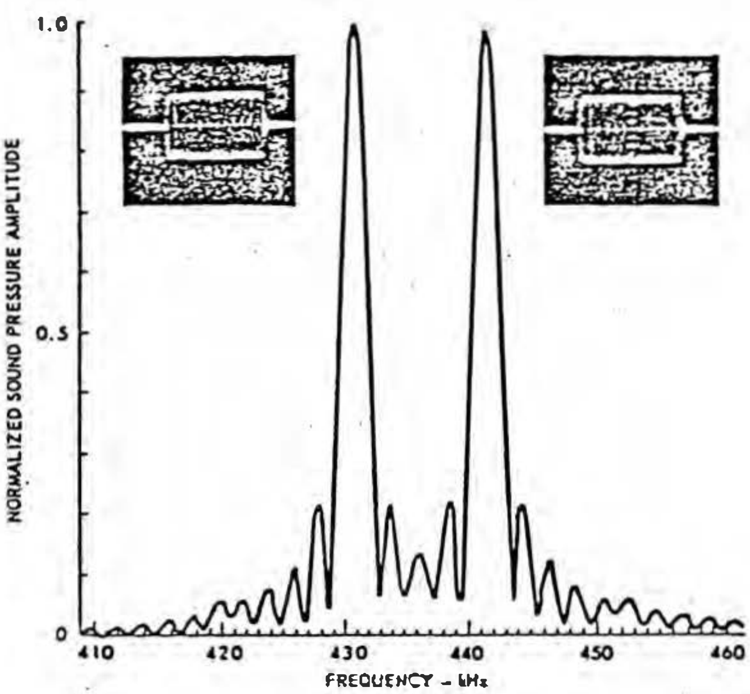
MULTIPLE SOUND PROPAGATION MECHANISMS

The solid materials in the reactor vessel allow shear wave as well as compressional wave propagation. This causes significant complication in interpretation of the returned vibrational energy.

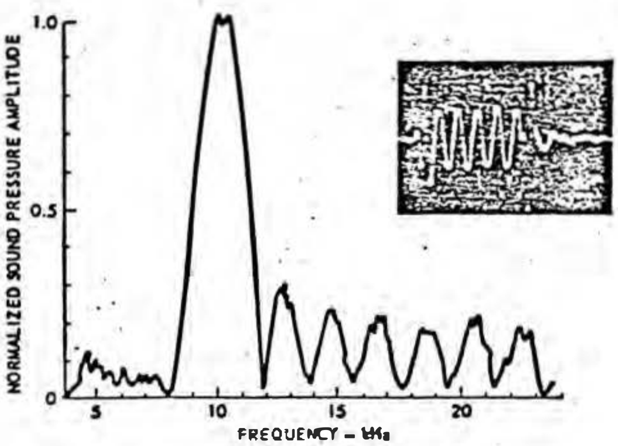
CONCLUSION

The highlighted factors combine to reduce the overall confidence that the acoustic imaging approach can provide the type of detailed engineering data desired to assist in material removal planning. The lack of previous studies on reactor core material, uncertainty of material mixture and configuration demand that experimentation be employed to evaluate the level of success achievable.

It is felt that further study of this problem will be unproductive without coordinated experiment. Benchmark material sound speed measurements in the overheated material from the fireburst facility would be the closest model known. A similar geometry proof-of-principal experiment using similar materials in a water-filled acoustic test tank to evaluate some of the complicating factors summarized above would assist in better defining how successful reactor material imaging might be. It is anticipated that to develop a workable measurement method several iterations of analysis and experiment would be required. The probable success rate estimate at this point without experimental evaluation must be rated as low. Cost estimates have been made for such a proof-of-principal experiment.



PULSE SHAPE AND FOURIER SPECTRUM OF THE CARRIERS



PULSE SHAPE AND FOURIER SPECTRUM OF DIFFERENCE FREQUENCY SOUND

TIME BANDWIDTH CHARACTERISTICS OF A PARAMETRIC ARRAY

FIGURE 1 Time Bandwidth Characteristics of a Parametric Array

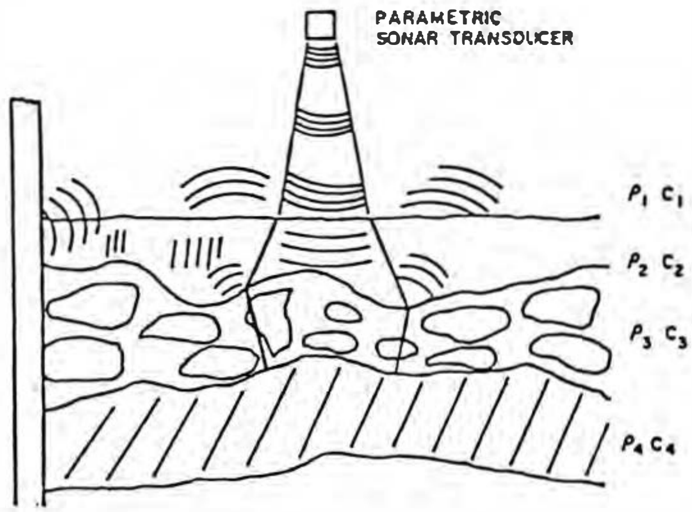


FIGURE 2a Complicating Factors in Acoustic Imaging Reactor Core Material

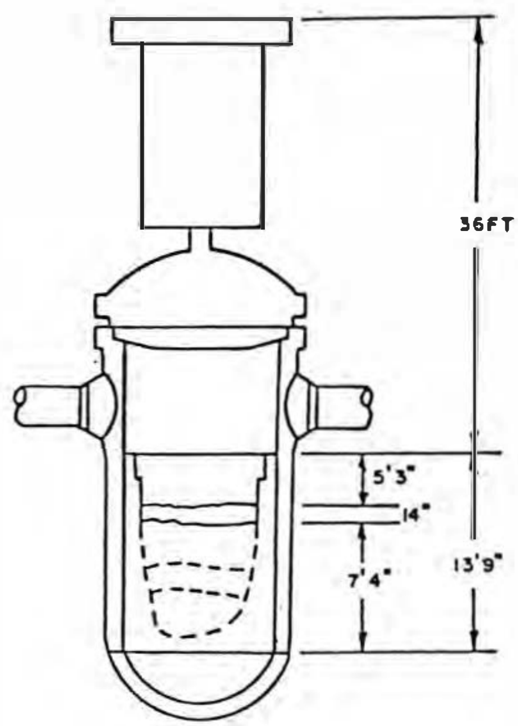
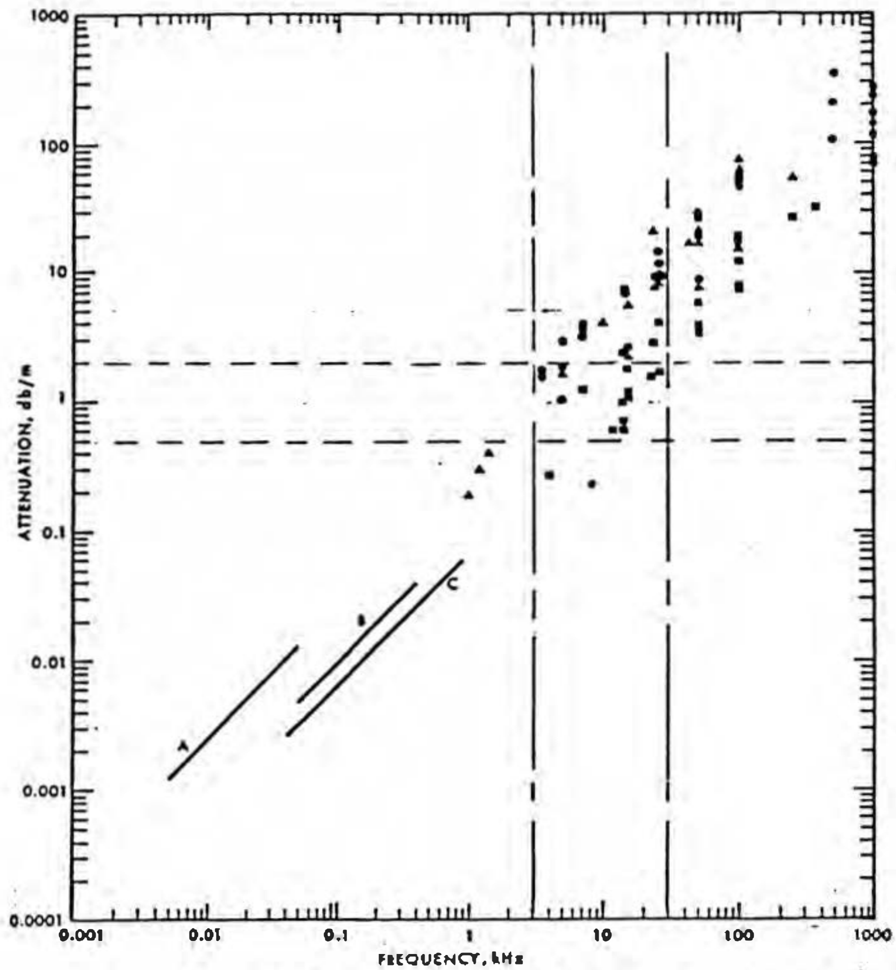


FIGURE 2b Reactor Dimensions



Attenuation versus frequency. The measurements are for natural saturated sediments and sedimentary strata: ●, sands (all grades); ■, clayey silt, silty clay; ▲, mixed sizes (e.g., silty sand, sandy silt, sand-silt-clay); sand data at 500 and 100 kHz. Low frequency data: line *A* land, sedimentary strata; line *B* Gulf of Mexico coastal clay-sand; line *C* sea floor, reflection technique. (Hamilton, 1972.)

FIGURE 3 Ocean Bottom and Land Sediment Attenuation Data

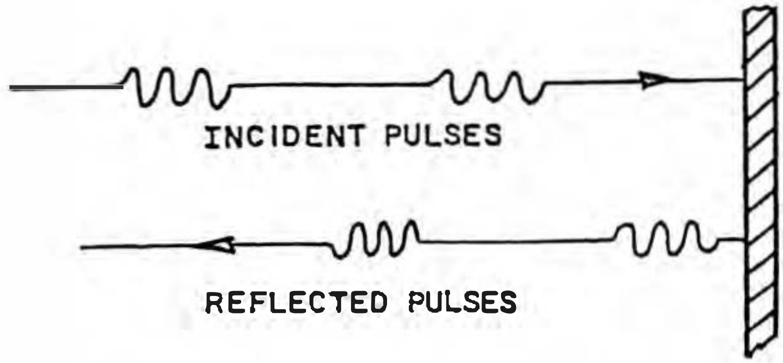


FIGURE 4a Reflection Resolution Equal to One-Half Wave Length. (Compressional Waves Depicted as Magnitude Traces for Concept Presentation Only.)

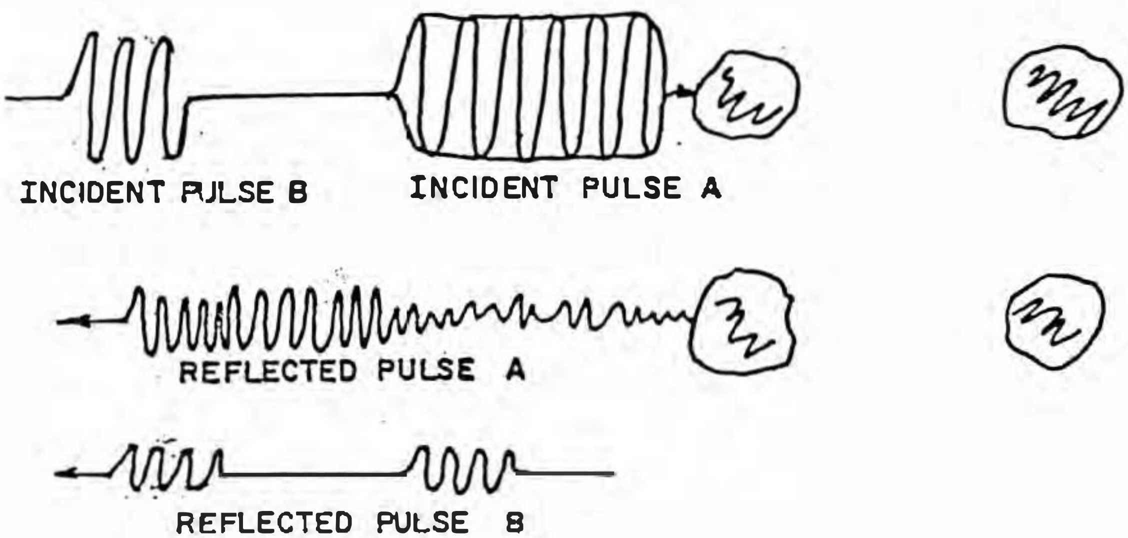


FIGURE 4b Example of Spacial Resolution for Two Closely Spaced Objects as a Function of Pulse Length.

MATERIAL PARAMETERS OF TMI CORE CONSTITUENTS

Material	Density ρ	Young's Mod. Y	Shear Modulus G	Poisson's Ratio μ	Bulk Modulus E	Compress. Speed c	Shear Speed c_s	ρc
Uranium Oxide UO_2	$10X10^3 \frac{kg}{M^3}$	$23.X10^{10} \frac{N}{N^2}$	$8.74X10^{10} \frac{N}{M^2}$	.316	$2.08X10^{11}$	$5700 \frac{M}{sec}$	$2960 \frac{M}{sec}$	$5.7X10^7 \frac{kg}{M^2\ sec}$
Zirconium Zr	$6.5X10^3$	$9.3X10^{10}$	$3.5X10^{10}$	.329	$9.06X10^{10}$	4600	2320	$2.99X10^7$
Zirconium Oxide	$5.0X10^3$	-	-	-				
Silver, Ind, Cad	$10.2X10^3$	$7.8X10^{10}$	$2.8X10^{10}$	.37	$1.00X10^{11}$	3670	1660	$3.74X10^7$
Inconel 718	$8.2X10^3$	$19.9X10^{10}$	$7.7X10^{10}$	.292	$1.59X10^{11}$	5650	3060	$4.64X10^7$
Stainless Steel 304	$7.8X10^3$	$19.3X10^{10}$	$7.6X10^{10}$	.262	$1.35X10^{11}$	5510	3120	$4.29X10^7$
Ceramic B4C	$3.45X10^3$	$44.9X10^{10}$	-	-	-			

GENERAL DYNAMICS
Electric Boat Division

00-01-1100 REV. 11/61 REF. SP 1-19

TELECON/CONFERENCE REPORT

☐ CONFERENCE ☐ TELECON

FILE	PAGE
	1

DATE & TIME OF MTG/TELECON: 5/13/83

	ACTIVITY/DEPT/CODE	LOCATION/ADDRESS	TELEPHONE NO
PERSON INITIATING DISCUSSION: J. Brown	EBDiv., Dept. 443	Groton, CT	(203)446-4600
PERSON CALLED: R. Grills	EBDiv., Dept. 732	Groton, CT	(203)446-2620
OTHER PARTICIPANTS: Richard Muenow	Muenow & Associates	Charlotte, NC	(704)377-4041

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JBrown:als
4600

APPENDIX IV-1

TECHNICAL MEMORANDUM

DATE 6/20/83

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TM3680-9, Rev. 1

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TO T. A. Hendrickson

FROM C W Hess
C. W. Hess

SUBJECT W.O. 3680-02
Technical Assistance and Advisory Group
Three Mile Island - Unit 2 Recovery
Defueling/Water Treatment Strategy

KEYWORDS TMI-2, Reactor, Fuel Debris, Water Filtration, Solids Handling

References:

1. "Third Report of Technical Assistance and Advisory Group (TAAG)", August 31, 1982
2. "Three Mile Island Unit 2 Core Status Summary: A Basis for Tool Development for Reactor Disassembly and Defueling", GEND 007, May 1981
3. "Pump Handbook", Igor Karassik et al, McGraw-Hill Books, 1976
4. "As Built Design and Material Characteristics of the TMI-II Core", B&W Report
5. "Engineering Properties of Ceramics," Battelle Memorial Institute, AFML-TR-66-52, June 1966
6. B&R Calculation #3680-21, "Defueling System Pressure Drop", 1/24/83
7. GPU Nuclear Planning Study, "Defueling Water Cleanup System", TPO/TMI-046, Rev. 0, April 1983
8. B&R Calculation #3680-24, "TMI-2 Water Cleanup Strategy", 4/7/83
9. B&R Calculation #3680-25, "Dose Rates from SDS Monitor Tanks and Piping", 4/11/83
10. "Proposal Methods for Defueling the TMI-2 Reactor Core", Debris Defueling Working Group, TC Runion Chairman, EG&G April 29, 1983

REFERENCES: (Cont'd)

11. B&R Technical Memorandum #TM3680-7, Rev.2, "Leach Rate of Activity into Reactor Building Basement Water," 4/4/83.

12. Telecon J.W. Walcott to C.W. Hess, "HPD Evaporator."

ATTACHMENTS

a. Figure 1 - Defueling/Water Cleanup System Schematic
b. Appendix 1 - Defueling/Water Cleanup System Design Criteria
c. Figure 2 - Particle Distribution
d. Table 1 - Fuel Debris Terminal Velocities
e. Figure 3 - Defueling Arrangements
f. Figure 4 - Typical Solids Handling Eductors
g. Figure 5 - Particle Settling Canister
h. Figure 6 - Cyclone Separating Canister
i. Table 2 - Reactor Vessel Cleanup Times
j. Figure 7 - Defueling System Arrangement
k. Figure 8 - Working Group Defueling System
l. Figure 9 - Cleanup System Flow Diagram
m. Table 3 - Water Cleanup System Valve Line Up
n. Figure 10 - GPUN Cleanup System
o. Figure 11 - GPUN Cleanup Filter
p. Figure 12 - Epicor II Flow Diagram

SUMMARY

An unusual aspect of fuel removal from the TMI-2 reactor will be the large quantity of fuel which is rubblized or broken into small pieces. 30% to 50% of the fuel is believed to be rubble and will be most effectively removed hydraulically by a "vacuum" system. This solids handling system's design is a potential critical path item for fuel removal.

A defueling/water treatment strategy is a prerequisite to the development of the defueling system design. This Technical Memorandum proposes such a strategy.

The proposed defueling/water treatment strategy can be summarized as follows:

1. The fuel removal vacuum system will operate independent of cleanup systems. The reactor vessel water will be separated from the fuel transfer canal water by a collar-type barrier extending from the top of the reactor vessel to the maximum elevation of refueling canal water.

2. Fuel debris removed by the vacuum system will be collected in small, disposable debris removal vessels and the sluice

water will be returned to the lower portions of the reactor vessel.

3. The size of the debris to be removed will decrease as defueling proceeds. Large fuel debris will be removed first by use of settling canisters, and, as necessary, smaller debris will be removed by utilizing cyclone separators or filters.

4. Hydraulic fuel removal will not require water clarity. The vacuum suction nozzle will be located "blindly," if necessary, and moved periodically without concern about visual location of suction nozzle.

5. Filled debris vessels will be stored in the fuel transfer canal until they can be loaded into the fuel shipping containers, or transferred to the "A" fuel storage pool in the failed fuel detection cans to be loaded into the shipping containers there. The vacuum system operation could proceed before the fuel shipping containers are available.

6. The reactor vessel water cleanup system will be utilized to maintain water clarity of the shield water. The system will filter the water, demineralize it, and return it to the "A" fuel pool. Clarified and demineralized water will flow back to the reactor through the fuel transfer tubes, into the fuel transfer canal, and back into the pool/reactor water barrier to the reactor vessel.

7. The vacuum system can be utilized to augment the water clarity system to restore water clarity near the rubble bed for the post-hydraulic defueling activities.

8. A disposable zeolite-based ion exchange system will be added to increase the ion exchange capacity to clean up any radionuclide spikes generated by defueling activities. This system will be available to augment the normal ion exchange system (SDS) or to replace it on a temporary basis.

9. Water used to flood the reactor building basement for shielding and to decontaminate by leaching can be processed by the Epicor II system. This system will not utilize the miscellaneous waste hold up tank (WDL-T-1) so that this processing can proceed independent of the miscellaneous waste system.

10. The TMI-2 miscellaneous radwaste liquids system will be available to process chemical decontamination waste solutions. A new radwaste evaporator and a portable solidification system will be required since TMI-2 used the TMI-1 evaporator and solidification system which probably cannot be made available for this effort.

A proposed Design Criteria for the various systems is given in Appendix 1. A schematic of the proposed defueling/water cleanup system is shown on Figure 1.

RECOMMENDATIONS

1. The reactor vessel water cleanup system should utilize, to the maximum extent possible, the spent fuel cooling filtration system modified as necessary. Reactor water can be drawn directly from the hot leg through existing decay heat removal piping, into the mini-decay heat removal system, then pumped directly to the spent fuel filters without any piping modifications.

   Minor modifications around the spent fuel filters to facilitate better filter handling and around the spent fuel demineralizer to permit the installation of the disposable zeolite ion exchanger will enhance the capabilities of this system to permit this duty.

2. The hydraulic defueling system should be designed to permit defueling without visual clarity. "Blind" defueling could be accomplished either manually or remotely, and would not require outages to restore water clarity.

3. The hydraulic defueling system should utilize a single debris removal vessel at a time which will remove predominantly one range of fuel debris size. This permits easier design, mock-up testing and defueling due to the simpler interface between the various defueling devices.

4. The bottom reactor vessel head (i.e., beneath the Core Support Assembly) should be surveyed visually prior to defueling activities in order to determine the amount of fuel debris that has collected there. If the quantity is significant, it should be the first area to be hydraulically defueled in order to address recriticality concerns. 1

5. All efforts should be made to minimize the amounts of water required for defueling. The water levels in the fuel transfer canal, the "A" fuel storage pool, and the reactor vessel barrier should be kept as low as possible to achieve the necessary shielding.

DISCUSSION

Strategy

Removing the fuel from the TMI-2 reactor is complicated by the large fraction of fuel debris. As much as fifty percent of the core, nearly 60 tons, is believed to be rubble. Two methods are

available to remove this fuel debris: mechanical or hydraulic. Due to the awkwardness of mechanical fuel removal approach, its man-REM cost for removing all of the fuel debris is likely to be prohibitive. TAAG has already recommended the use of a "vacuum" system to remove a large portion of the fuel debris. (Ref. 1)

That report recommended an integrated system that would remove fuel, package fuel, clarify the shield water, and demineralize all process water to remove fission products, especially cesium and strontium. This approach had several advantages: 1) it would clean up the water near the probable location of contaminant generation, 2) outbreaks of cesium from disturbed rubble beds would be prevented from becoming generalized radiation problems, and 3) a single system would be utilized to accomplish all defueling and cleanup activities. However, in order to develop such a system, all the system design requirements and interfaces must be known. Each portion of the system is dependent on the other portions. If suitable redundancy is not available, an outage of any subsystem will necessitate the termination of defueling or cleanup activities.

The actual design difficulties are compounded by the lack of
knowledge about the size distribution, the density, and the com-
position of the rubble. The determination of the size distribu-
tion is a critical path item for the system design. No reactor 1
vessel fuel debris samples have been taken and there is no
assurance that any sample contemplated would be representative.
Hence, the most important design data, the characterization of
the fuel debris, is not available. It is questionable whether it
is possible to design a system to remove solids without basic
physical data about the solids. Due to the high specific acti-
vity of fuel debris, installing a complicated system based on
assumptions is not acceptable.

In order to separate the design problem into discrete subgroups, the defueling/water treatment strategy proposed in Ref. 1 must be revised. The three major aspects of the previous approach, i.e., fuel removal, reactor vessel water filtration, and water cleanup, will be retained. However, the defueling function will be separated from the water cleanup functions. This permits the following advantages:

1. Since the effluent of the vacuum system is not sent to the water cleanup systems, it is not necessary to remove all of the fuel debris in the debris vessel. Hence, rapid filter clogging resulting in nearly empty debris vessels can be avoided. Returning the effluent water to the reactor vessel will keep the fuel in the "pot" until it can be collected.

2. The early assessment of the fuel debris size distribution is not required since the defueling system will not attempt to remove all debris in one pass. 1

3. The water cleanup subsystems are no longer tied to the defueling system. Hence, their flow rates need not be determined by defueling requirements, and their operation need not be constrained by the defueling system's operation.

4. It is now possible to utilize existing systems for the water cleanup systems since the special requirements of the defueling system no longer control the design.

5. The water cleanup systems can remain on line for the mechanical fuel removal phase of the defueling. Thus, modifications to the cleanup systems need not be critical path items for later defueling efforts.

This proposed defueling/water treatment strategy does have limitations. The water clarity inside the reactor vessel may not be adequate to maintain visibility during defueling activities, and the contamination of the reactor coolant may result in high dose rates on the defueling work platforms. These limitations are addressed by the following:

1. The placement of the suction nozzle will not require the operator to see the nozzle.

2. A reactor vessel/fuel canal barrier will prevent fission product contamination from becoming a generalized effect in the fuel transfer area.

3. If dose rates are too high, the polar crane or other remote lifting device can be used to place the suction nozzle. Also, the water cleanup system is focused on the shield water and will work to mitigate dose rate spikes.

Hence, this proposed strategy commits to "blind" suction nozzle placement. It should be noted that this is probably the realistic approach regardless of intent. If the cleanup system takes suction from one of the reactor vessel nozzles and returns clean water to the top of the reactor vessel barrier, it will be ineffective in cleaning fuel fines in the area of the fuel debris. (See Figure 1.) If a manifold is lowered to the vicinity of the fuel debris in order to mitigate this problem, it could result in inadvertent fuel removal by the water cleanup system. Thus, it is likely that the only system to maintain water clarity near the fuel will be the "vacuum" system.

Additional steps can be taken to reduce the problems of fuel fines and water clarity near the fuel. For instance, the veloc-

ity of the fluid in the suction piping of the vacuum system must be kept high to assure fuel transport up to the debris filter. However, the return fluid velocity need not be high. Hence, by design, the vacuum system could return low velocity water to the bottom of the reactor vessel (i.e., beneath the core basket). This would reduce the probability of colloidal sized fuel fines becoming widely distributed by developing a circulation cell low in the vessel.

Fuel Removal System

General:

The "vacuum" system is a hydraulic dredging system designed around the size and character of uranium dioxide fuel debris. The size range is from micron sized fines to intact fuels pellets. (See Figure 2.) Following standard practice for such systems, the inside diameter of the piping system should be 2.5 to 3 times the dimension of the largest solid to be transported. (Ref. 3.) An intact fuel pellet ought to represent the largest object to be handled by the fuel removal system on a routine basis. (Ref. 4.) An intact pellet at TMI-2 is .7" long with a diameter of .37" (Ref. 4) which implies that the fuel removal piping system should employ a 1.75" to 2.1" internal diameter.

Table 1 shows the range of fluid velocities required to lift various sized fuel debris (assumed to be spheres) and pellets. It can be seen that the flow rate required to lift a wide range of particle sizes does not vary widely. The fact that the fluid velocity required to remove intact pellets is not very different from the velocity required to remove large debris means that it will not be possible to selectively discriminate against fuel pellets by controlling the flow rate.

The defueling system should be located inside of the reactor vessel barrier. Since it will be necessary to make and break connections to vessels, pumps, and valves, it is logical to keep all of the resultant fuel debris releases contained inside of the contaminated area. In order to utilize the reactor internals indexing fixture, a shielded transfer bell should be developed to facilitate the transfer of the filled debris vessels out of the R.V. barrier to a designated storage area in the fuel transfer canal. An internal wash down header should be integrated into the transfer bell design so as to remove as much contamination as possible before transferring the vessels out of the barrier.

Piping:

The piping system for the fuel removal system will be exposed to a harsh environment. Of prime concern are the mechanical properties of uranium dioxide (UO_2) fuel debris. The fuel used at

TMI-2 has a 92.5% of theoretical density, or about 633 pounds per cubic foot (10.14 gm/cm^3) (Ref. 4), and a hardness of 6-7 Mohs (Ref. 5). This is comparable to pumping silica sand with the density of lead. Clearly, the piping in contact with the fuel debris must be designed to minimize wear. Velocities must be kept as low as possible and bends should be as gradual as practical. The piping must also be abrasion resistant and flexible. These requirements all support the use of abrasion resistant rubber hoses instead of steel or alloy piping with the anticipation of routine replacement of hoses to compensate for wear. In order to keep the fluid velocities as low as practical, horizontal runs must be minimized or plugging will result.

Pump:

Figure 3 shows six system design concepts for the fuel removal system. These concepts are variations of typical arrangements for hydraulic dredging systems to reflect the special needs of defueling. Concept A is a straightforward system consisting of a suction line, a slurry pump, a solids removing device, and a diffusing return line. The advantages of this system are as follows:

1. Water flow path minimizes turbulence in water above fuel debris.

2. Full pump pressure is available to drive water through the debris vessel.

3. No NPSH problems for the pump.

However, this arrangement does require the pump to be exposed to the full range of fuel debris. In view of the wide range of debris size and of the abrasiveness of the UO_2 fuel debris, this would impose severe restrictions on the type of pump that could be used.

There are two general approaches to selecting the materials for such pumps (Ref. 2): 1) soft-lined or elastomer parts, or 2) hardened abrasion resistant parts. Soft pumps are best suited for slurries with uniform, small sized particles, smaller in size than Mesh 7 (less than .11 inch). Large particles tend to cut the elastomer pieces and reduce the useful service life of the pumps. Hardened pumps have practical metallurgical limits to the hardness they can achieve. Since a Mohs index of 7 is approximately equal to a Birnell Hardness Number (BHN) of 650, the UO_2 fuel debris will be of a hardness equivalent to the maximum hardness possible for metals (Martenistic White iron such as Ni Hard, or 15/3 alloy). Hence, it is likely that Concept A will require a severe service pump which cannot be expected to operate reliably without maintenance.

Concept B on Figure 3 is similar to Concept A except that the pump is placed downstream of the debris vessel. This arrangement reduces both the amount of and the size of the solids which pass through the pump. This would permit the use of the less expensive "soft" pumps or would extend the useful life of a "hard" pump. Unfortunately, many "soft" pumps cannot tolerate low suction pressures because the liners separate from the base metal. Hence, a pump with hard wetted parts (BHN $\geq$650) would be required. A disadvantage of this concept is the practical limit on the pressure drop through the debris vessel device of about 10 psid. This can be improved somewhat by submerging the pump to take advantage of the pressure of the shield water. However, that solution increases the difficulty of performing any required maintenance that may be required.

Concepts C and D on Figure 3 eliminate all solids requirements for centrifugal pump by removing the pump from the slurry flow stream. This is done by the use of a small eductor to pump the slurry. The maximum flow rate required for this system is approximately 120 gpm which is well within the operating range of small (2" to 4") eductors. The centrifugal pump would take suction from the shield water so it would not need to be a solids-handling pump. Also, since the shield water will not be particularly radioactive, the pump can be located in an easily accessible area to facilitate operation and routine maintenance. The difference between Concept C and D, the location of the debris vessel, is a function of the specific performance characteristics of a given eductor. Clearly, placing the eductor downstream of the debris vessel will increase the service life of the eductor. This results in the same limitation on the debris vessel as does Concept B, i.e., maximum pressure differential of ~ 10 psid. However, placing the eductor upstream of the debris vessel, as in Concept C, raises the potential for blockage due to the restriction of the diffuser portion of the eductor as well as reintroduces the problems of abrasive slurry handling and shortened operating life. It also will reduce the effectiveness of a settling cannister by increasing the flow rate through the vessel.

Concept D has an additional advantage in the ease with which a back-wash flow can be established. Deadheading an operating eductor causes the motive fluid to be diverted backwards down the suction line. This reverse flow of clean water can be used to backflush the debris vessel and the suction line. No other concept has such a straightforward back-wash capability. Concept C can easily back-flush the suction line but cannot backwash the debris vessel.

Concept E on Figure 3 shows the use of a solids eductor lowered to the rubble or a specialized dredging eductor called a sand and mud eductor. (See Figure 4.) These eductors would be required

if the pressure drop of solids transport up the line is too great to be generated by a pump. A sand and mud eductor is specially designed to remove silt and sand from rubble beds. High velocity water jets are directed at the rubble and the eductor takes suction from the resultant cloud of suspended solids around the bottom of the eductor. This prevents clogging of the eductor because the suction nozzle does not touch the rubble bed. Also, since the fuel fines are so dense, this approach may permit the discrimination between large dense fuel debris, and fines or other lightweight fuel debris, such as zirconium cladding. Of course, such a device may result in the suspension of a large amount of fuel debris which would decrease visibility around the dredging site more than necessary. A sand and mud eductor will not be effective in removing either coarse fuel debris or intact pellets.

Concept F on Figure 3 places the pump downstream of the solids eductor. Such an arrangement would be able to deliver the discharge head from the pump to secondary debris vessels downstream of the primary debris vessel. Such a setup has advantages if numerous debris vessels are to be used in series. The use of the solids eductor near the rubble bed provides enough pressure to the suction of the pump to permit the use of a primary debris vessel with a pressure drop greater than 10 psid. The centrifugal pump could probably be a "soft" type pump due to this supplied pressure, thus reducing the cost of the pump. Unfortunately, the pump would be in the debris flow path and will need to be shielded.

A disadvantage of Concepts C, D, and E on Figure 3 is that they all set up an undesirable circulation path within the reactor vessel. Depending on the suction lift, the amount of water needed to drive the eductor can be nearly twice the suction flow rate. This will result in a net flow upward from the rubble bed which could carry fuel fines into the shield water. This effect will be the most prominent with Concept D due to the large suction lift requirements imposed by the solids removal device. Concept E will minimize this effect because both a smaller hose and a lower fluid velocity are possible because of the smaller sized fuel debris.

Concept D is selected as the best concept for use as the coarse debris defueling system. It has no moving parts in the slurry, so that wear and maintenance can be minimized. The eductor will be in a low abrasive environment and need not be specialized for solids-handling service. The eductor is small enough and inexpensive enough to permit disposal if it should fail. Replacement eductors could be maintained near the defueling operating area to minimize outage time for eductor replacement. The centrifugal, self-priming pump can be an inexpensive unshielded conventional pump with a standard electrical motor (no need for submersible

motors). If desired, its suction point can be high enough to clean any floating debris from the shield water. A full-ported, ball valve can be installed downstream of the eductor to provide back-wash capability.

The choice of Concept D impacts both the debris vessel and the water cleanup system design. The practical limits of suction pressure require that the debris vessel be limited in differential pressure to about 10 psid. The upwelling of fuel debris from the rubble bed will have to be intercepted and removed by the water cleanup system in order to maintain a water quality of the shield water.

Once the coarse material is removed, Concept C or E might be required to drive the water through devices with larger pressure drops such as cyclone separators or filters. However, both of these systems are simple rearrangements of Concept D and will not take much time or effort to put on line.

Debris Vessel:

There are three (3) general types of non-active devices which could be used to collect the fuel debris: 1) filters, 2) settling canisters, and 3) cyclone separator canisters.

Filters are the most commonly used device for removing particulates from liquid effluent streams. They are best suited for use in streams with low weight percent solids. Since the fuel debris will be transported in high weight percent slurries, filters are not suited for use as the primary solids removal device. A filter may be used downstream of another debris collection vessels to prevent fines from being returned to the reactor vessel. Such a filter would need to be on the discharge side of the pump (either centrifugal or eductor) due to the high pressure drops associated with small, micron-sized filters. The loading efficiency of a disposable micron filter will not be very high even under the best conditions. This will result in a large number of filter vessels being required with all the attendant system outage time for replacement. A back-flushable filter, if it is carefully engineered and designed, may reduce this liability but at the cost of lead time and increased cost.

Particle settling canisters operate by reducing the fluid velocity below the limit required to keep the particles fluidized in the stream. (See Figure 5.) Due to the size limitations imposed by the fuel shipping casks, the canisters cannot be much larger than 12" OD. This means that the maximum reduction fluid velocity is about a factor of ten, i.e., from about 10 fps to less than 1 fps at 150 gpm. From Table 1 it is clear that such a device will be adequate to remove fuel debris above .010" in nominal diameter. According to Ref. 2, this represents 20-60% of

all fuel debris. (See Figure 2.) The amount of fuel collected in the settling canister can be raised by placing a backwashable filter element on the discharge line. However, such a filter element would cause operational problems by requiring frequent backwashing to clear blockages, especially if its mesh is too small. Due to these limitations, it is probable that the filter element should be avoided and that the settling cannister should be used for large fuel debris only.

A cyclone separator operates by taking advantage of centrifugal separation of denser solid fractions from the transporting liquid. (See Figure 6.) In general, small cyclones work best to remove small particles, but, obviously, cannot handle large particles. Large cyclones can handle large particles but do not remove small particles with the required efficiency. Due to the variability in the sizes of particles to be handled, cyclone separators are not good choices to process the entire defueling slurry. Either some other device which will eliminate all debris above a given size (say 500 microns) must be placed upstream of the cyclone, or the cyclone can only be used when the expected fuel debris size is small. The cyclone can either be an integral part of the canister, and therefore disposable, or the cyclone could be a major piece of equipment which empties into a disposable container. The first approach has the advantage of not requiring a specialized coupling with respect to the other types of solids removal devices, hence the cyclones can be interchangeable with other devices. The second approach permits tighter stacking of the canisters in the failed fuel shipping containers, hence fewer shipments. In order to reduce wear, the cyclones should have an involuted feed rather than the usual tangential feed.

Whichever debris vessel design is used, care must be taken to prevent recriticality. The K effective of the assembled fuel shipping containers must be less than .95 by design. Each vessel must have dewatering capability to reduce radiolytic generation of gases as well as to enhance the shippability of the vessel.

It is recommended that each solids removal vessel be short, three to four feet in length, to avoid the need for deep water shields. This will also permit the use of an inexpensive, shielded transfer bell to be used to transfer the canisters to the deep end of the refueling canal for storage or for transfer to the "A" storage pool. Reducing the shield water depth requirements will minimize the amount of water required in the refueling canal which will improve the cleanup rate for the canal water.

It is clear from the preceding discussion, that no single debris vessel design will be adequate for all defueling conditions. Settling canisters are best with a wide range of large solids. Cyclones are best with a low weight percent solids slurry of

small particle size. Filters are best for cleanup systems and for final debris removal.

The layout of the proposed defueling strategy (showing pump concept D from Figure 3, with a single defueling vessel) is shown in Figure 7.

Debris Defueling Working Group Proposal:

The system proposed by the debris defueling working group report (Ref. 10) is essentially identical to the approach first recommended by TAAG (Ref. 1). It consists of a series of debris removal devices, each specializing in a specific range of particle sizes, and incorporates a feed stream to SDS for cleanup. Since this approach requires knowledge of the size and density distributions of the debris, the working group spent a great deal of effort trying to define these parameters. While the results are illuminating, there is no indication that their assumed debris characteristics are representative of the TMI-2 rubble. Thus, the reservation, stated previously in this Technical Memorandum, that a realistic system cannot be designed on assumed debris characteristics, has not been changed by the working group's findings. This is especially true because the working group did not consider data being obtained from the Power Burst Facility failed fuel experiments about the nature and character of fuel debris. Thus, it still seems expedient to attack the debris removal phase piecewise, removing one type of debris at a time and returning the other debris back to the reactor vessel.

The working group assumed that water clarity was needed for hydraulic defueling and recommended steps be taken to minimize turbidity near the rubble bed. If adhered to, such an approach would require termination of the defueling effort each time the water clarity degraded. Thus, much time which could be used to defuel the reactor would be wasted trying to restore water clarity which is, at best, a secondary goal. It is important to design the hydraulic defueling system to operate without the need for water clarity in order to minimize the amount of time and effort spent in high radiation areas to remove the debris.

The working group's system (see Figure 8) includes filtration and demineralization. This seems to duplicate the function of the cleanup system proposed by GPU Nuclear. Hence, it would seem reasonable to modify either the water cleanup system or the Working Group's system to remove this redundancy.

The working group's hardware recommendations are useful and compatible with the strategy proposed in this memorandum. The only difference is that B&R's defueling strategy would use each type of debris removal device individually and sequentially rather than attempt to operate them in series.

Reactor Vessel Cleanup System

General:

The major purpose of this system is to provide the capability to maintain the water clarity of the shield water in the reactor vessel. A secondary purpose will be to control the cesium and other radionuclide concentrations in the shield water to keep dose rates as low as practical above the vessel. Two approaches are possible: 1) design and construct a new system, or 2) utilize existing systems modified to suit the defueling requirements. The second approach is clearly superior if the modifications required are limited and result in a functional system.

The major parameters required to size a cleanup system are the quantity of water to be processed and the cleanness required. The strategy proposed by this Technical Memorandum does not require water cleanup during much of the defueling activities because of the "blind" defueling approach. Also, water volumes to be processed shall be minimized. Thus, the size of the system required by this strategy is smaller than might otherwise be the case.

Another design consideration is the flow pattern to be established in the reactor vessel. If the flow path creates significant fluid velocities across the rubble bed, it may fluidize fuel debris and result in rapid loading of the filters. As already discussed, filters are not ideally suited for high weight percent solids slurries. Hence, such a flow pattern should be avoided. This means that the fluid flow path of the cleanup system should not impinge on the rubble bed. Since no cleanup flow will exist near the rubble bed, defueling suction nozzle will likely be obscured regardless of the cleanup flow rate. This is the major rationale supporting a "blind" defueling strategy.

Existing Systems:

After a review of the existing systems in communication with the reactor coolant system, it is clear that the spent fuel cooling filters (SF-F-1A and 1B) could serve as the cleanup filters if the flow rate of the cleanup system can be kept below 200 gpm. Table 2 shows the sorts of cleanup times required for a range of water volumes at a range of process flow rates. It can be seen that 200 gpm is adequate if the water volume can be kept below 35,000 gallons. This is possible if the shield water level is kept below EL. 328'6".

The normal flow path to the spent fuel filters from the reactor coolant system is established by the operation of the decay heat

removal system. This flow comes from the decay heat removal system piping downstream of the heat exchangers DH-C-1A and 1B through valves DH-V106A and 106B. (See Figure 9.) This reactor coolant flow can be directed through either of the spent fuel filters or through the spent fuel demineralizer and returned to either the spent fuel pools or to the reactor coolant system directly. Although this process path is ideal, the flow rate of the decay heat removal pumps (3000 gpm) is too large to be practical. Also, this arrangement would place the entire decay heat removal system in the process path which is not desirable.

The mini-decay heat removal system was installed after the accident and is in parallel with the existing decay heat removal system. This small system takes suction from the decay heat removal letdown line (downstream of valve DH-V3) and discharges back to the decay heat removal return line (upstream of valve DH-V4B). Utilizing existing piping and pumps, the mini-decay heat removal system could be used to pump 130 gpm of water through the spent fuel filters. (Reference 6.) Table 3 is a valve lineup for this flow path which bypasses the mini-decay heat removal heat exchangers. Using the largest impeller which can fit in the existing MDHR pumps (8"), this flow rate can be raised to 170 gpm. Utilizing the borated water recirc. pump (SF-P-2) as a booster pump, the flow from the combined system is limited to runout flow rate of the MDHR pump, 170 gpm for the existing configuration and 210 gpm for the 8" impeller.

The fluid velocity in the 8" lines at these flow rates will be .8 to 1.4 fps. This is not adequate to keep large particles suspended in the slurry. However, the approach velocity to the hot leg, from which this system takes suction, will be less than .07 fps which means that very few large particles will be ingested. Thus, this liability will be minimized.

An alternative to using the mini-decay heat removal system would be to install a new system in the fuel handling building basement to tie together the decay heat removal system and the spent fuel cooling system. Such a system would have numerous advantages over the use of the mini-decay heat removal system: 1

- Minimize contamination of existing piping systems
- Shorter piping runs
- Smaller piping sizes to keep velocities high
- Control over crud traps
- Built-in decontamination capability
- Flow rate can be easily varied.

This approach maintains many of the advantages of the proposed concept without the disadvantages. The cost would obviously be higher for this alternative but it would still represent a significant savings over the totally new system in containment approach.

In addition, most of this construction work would take place in low dose rate areas and would represent a minimal man-REM cost. 1

The spent fuel filters are suitable for the reactor vessel cleanup system as long as the weight percent solids in the slurry and the amount of UO_2 in the solids can be minimized. These conditions can be met by the selection of the suction point and the "blind" defueling approach. Obviously, the shield water cleanup is the only important water body to be kept clean. The shield water cannot be a significant source or its purpose is negated. Thus, the cleanup system must maintain low levels of solids and dissolved radionuclides in order to permit access to the defueling platform. Since the maximum flow rate upward from the rubble bed will be 4.0E-3 fps, very few fuel fines should be transported into the shield water. Lighter weight fuel debris, such as cladding debris and corrosion products, may be transported upward but it will not be highly radioactive. Any credible outbreak of soluble radionuclides, most notably cesium, from the disturbed rubble bed (i.e., a factor of 10 increase in the present release rate of 2.0 Ci/day) can be easily handled by the reactor vessel cleanup system flow rate.

The installed spent fuel cooling system demineralizer (SF-K-1) is not adequate for use as reactor vessel cleanup system demineralizer mainly because it is based on organic ion exchange resins. Also, the resin transfer system sluices spent resin from it to TMI-1, which is undesirable. However, ample room exists outside of the spent fuel demineralizer cubicle to place a full-flow, disposable, zeolite, ion exchanger for cesium spikes. The tie-ins would be such that existing piping and valves could be utilized. Hence, the temporary ion exchanger could be put on line from the control room to clean up spikes of activity or to quickly demineralize the spent fuel pool and fuel transfer canal.

Normal demineralizing could be accomplished by the submerged demineralizer system (SDS) located in the "B" fuel pool. Filtered and demineralized flow from this reactor vessel cleanup system would be discharged into the "A" fuel pool. It would flow to the fuel transfer canal through an open transfer tube, and would be siphoned back into the reactor vessel over the top of internals indexing fixture. Thus, the flow path of the reactor vessel cleanup system would circulate water throughout all of the important water bodies and would maintain a flow toward the areas of higher contamination.

The major advantages of utilizing the existing plant systems for the reactor vessel cleanup system are:

- Low cost
- Operated from control room
- No man-Rem associated with construction

- o Low man-Rem associated with start-up/mods
- o No adverse impact on schedule

There are, of course, disadvantages to this approach. First, the maximum water level in the "A" spent fuel pool and the fuel transfer canal is limited to elevation 328' (6" of freeboard on indexing fixture). This means that the plenum assembly will need to be stored in the deep end of the fuel transfer canal to provide the necessary shielding. This is possible but it does eliminate much of the useful storage area for filled defueling canisters. This may force the transfer of defueling canisters into fuel shipping containers soon after filling the canisters. This approach means that no defueling can take place until the fuel shipping containers are available. This is not desirable due to the potential delays in design, construction, and/or delivery of the shipping containers. If a solution to the fuel shipping containers is not forthcoming, the defueling canisters can be either transferred to the "A" fuel storage pool by use of a temporary transfer rack where they would be eventually transferred into shipping containers or the defueling reactor vessel barrier can be extended upward to permit more shield water and the storage of the defueling canisters in the fuel transfer canal until the shipping containers.

Another disadvantage of this approach is the coupling of the reactor vessel and the spent fuel pool/fuel transfer canal cleanup systems. The only cleanup flow for the "A" spent fuel pool and for the fuel transfer canal is the effluent water from the reactor vessel cleanup system. The reactor vessel cleanup system flow rate is not large enough to effect rapid cleanup of the large water volumes involved. Also, since effluent water is used, any breakdown in the system will spread contamination into clean areas. However, it should be noted that the reactor vessel cleanup system's flow rate is about the same as the normal spent fuel cleanup system flow rate and that the volume of water to be cleaned is only 38% of the normal defueling water volume (400,000 vs. 1,050,000 gallons). Hence, the cleanup rate is actually quicker than normal. In addition, the reactor vessel cleanup path can be isolated from the pool/canal cleanup flow path by use of the canal drain line and the borated water recirculation pump (SF-P-2). This will circulate the total water volume every 33 hours.

Other disadvantages, such as low water level, reducing shielding, and the lack of fuel pool cooling systems, are generic to most approaches and mitigated by the relatively low activity and heat generation rate for this fuel debris.

GPU Nuclear Proposal:

GPU site personnel have developed a defueling water cleanup system which has many of the same features of the system proposed in this Technical Memorandum (Ref. 7). A schematic of that system is shown on Figure 10. This design is the result of a different approach to the problem from the one taken by this Technical Memorandum. Whereas this Technical Memorandum began with existing systems and devised a strategy based on their performance, GPU Nuclear began with the maximum volumes of water that might need to be processed and designed systems based on postulated events during defueling.

The GPU Nuclear reactor vessel cleanup system is a 400 gpm filtration system with a 20-60 gpm zeolite ion exchange cleanup stream. The size of these streams was based largely on the amount of water assumed to be present, 78,000 gallons. Based on the calculation performed for this Technical Memorandum (Ref. 8), this value is too large. The reactor vessel with the head off, the upper plenum assembly removed, but with all other internals in place holds less than 28,000 gallons. A reactor vessel barrier with a diameter of 172" (equal to the internals indexing fixture) filled to elevation 347'6" contains only 30,200 gallons. Hence the maximum amount of water in the system is 58,200 gallons. In addition, not all of this water will be processed due to the flow path established by the cleanup system, which will take suction from just above the debris bed, ~E305'. This results in a further reduction in the amount of water to be processed of approximately 5,000 to 10,000 gallons. A better more realistic estimate of the water volume is ~50,000 gallons which is 36% less than GPU used. For equal concentrations of contamination, this would permit the use of a 300 gpm system in place of the proposed 400 gpm system.

However, even this amount of water is inappropriately large because it is unlikely that good water clarity will result because of the unwieldy flow path. Unless a suction nozzle is lowered to the vicinity of the rubble bed, there will be little cleanup of the area near the defueling activities. Since 400 gpm is four times the flow required to remove the fuel debris, care must be taken to prevent gross fuel removal. In order to prevent the water cleanup system from acting as a defueling system, the water velocity must be less than ~.1 fps, or the suction nozzle must be some distance removed from the rubble bed. At 400 gpm, a 40" ID nozzle is required to keep the fluid velocity below .1 fps. Alternatively, a manifold of nozzles could be used but would be an unwieldy device. This means that the suction nozzle for the R.V. cleanup system must be removed from the area of the rubble bed in order to prevent gross defueling. Hence, the cleanup of the water immediately above the rubble bed will not be accomplished by the R.V. cleanup system even at 400 gpm.

Also, the flow path will do little to remove floating debris from the top of the shield water. It is credible that there could be finely divided particles which will float or remain suspended in the water near the top of the defueling barrier. It is unlikely that returning filtered water to the top of the barrier will successfully remove this floating debris.

The proposed filter design for the R.V. cleanup system is a porous metal tube bundle with a drop-out canister. The flow is through the tubes, inside to outside, with any large debris passing straight through to the drop-out canister (see Figure 11). Such a design must of necessity utilize very large tube surface areas. The velocity of approach must be low to permit the majority of the solids to carry through the tubes to the canister below. For the smaller debris, the major phenomenon is filtration; hence, to avoid the need for frequent backwashing, large tubes are required. Sizing the filtration elements will be a major design effort requiring mockup testing. In addition, the system will not back flush effectively. Since the drop-out canister has no flow, the back washed solids will be flushed back into the reactor vessel. With the maximum canister diameter being held to 12", the 400 gpm flow rate seems too large to accommodate this design and will necessitate a multi-unit filter train manifolded together. While not unworkable, such a system will be cumbersome and expensive, and will require a long lead time for design and procurement.

The size of the GPU Nuclear system is compatible with use of auxiliary building systems if the hot leg suction path is used. However, GPU Nuclear does not plan to utilize this approach. Instead, large portions of the R.V. cleanup system are to be built and installed in the reactor building. The man-REM cost of such an approach is certain to be high. There are very few reactor building penetrations available for use by this system so that it will either be built entirely inside containment or that costly piping system designs will be necessary.

The GPU Nuclear system does have separate systems for R.V. cleanup and for fuel pool and transfer canal cleanup. This has some advantages over the method proposed by this Technical Memorandum. However, the driving rationale for this decision was the fact that zeolite ion exchange media requires a fairly constant influent concentration. If the concentration drops dramatically, as would be the case if the same ion exchanger was switched from reactor coolant chemistry to spent fuel pool chemistry water, the low concentration water will elute ions off of the zeolites resulting in increased contamination of the water. However, such a concern could be addressed by the use of two separate ion exchange systems, one for normal cleanup duty, and one for spikes or surges. The approach taken by this Technical Memorandum is to utilize SDS for the normal cleanup demineralizer

system and a new temporary ion exchanger replacing SF-K-1 for the transient loads.

Other Waste Cleanup Systems

Reactor Building Basement:

If current radcon engineering studies of the basement lead to the conclusion that the basement should be reflooded for shielding as well as to remove leachable cesium, a water cleanup system must be made available to keep this shield water below 1μCi/ml. At the current cesium appearance rate in the basement of 25 Ci/day (Reference 11), only 5 gpm at 1μCi/ml is required to maintain the equilibrium. Thus, this cleanup can be accomplished with a relatively low flow rate system and is well within the capabilities of the Epicor II system.

Figure 12 shows a potential flow path to use Epicor II as the demineralizer cleanup system for the basement shield water. This system has the following advantages:

- o Extensive use of existing systems
- o Limited electrical requirements
- o Frees the Miscellaneous Waste Holdup Tank
- o Low cost
- o Low man-REM
- o Does not impact existing systems operability

The use of the SDS monitor tanks (SDS-T-1A/1B) provides both easy tie-ins and surge capacity to accommodate the normal Epicor II outage rate. It also separates the miscellaneous waste holdup tank (WDL-T-1) from Epicor II. Shielding analysis of a monitor tank filled with 1μCi/ml water indicates that the dose rates will be on the order of 25 mR/hr (Ref. 9). Hence, dose rates near the tanks need not be a major concern.

The Epicor II resins will need to be changed to optimize the system's characteristics for this service. Other modifications to the Epicor II system would be necessary to restore it to its original cleanup system configuration.

Decon Solutions:

Later in the decontamination efforts, harsh chemicals may be required. Such chemicals are not compatible with ion exchangers for cleanup. A radwaste evaporator will be necessary to reduce the volume of this waste for disposal processing.

The normal process path for such wastes at TMI-2 is the miscellaneous radwaste liquids disposal system. This system includes neutralizing capability to handle both caustic and acidic

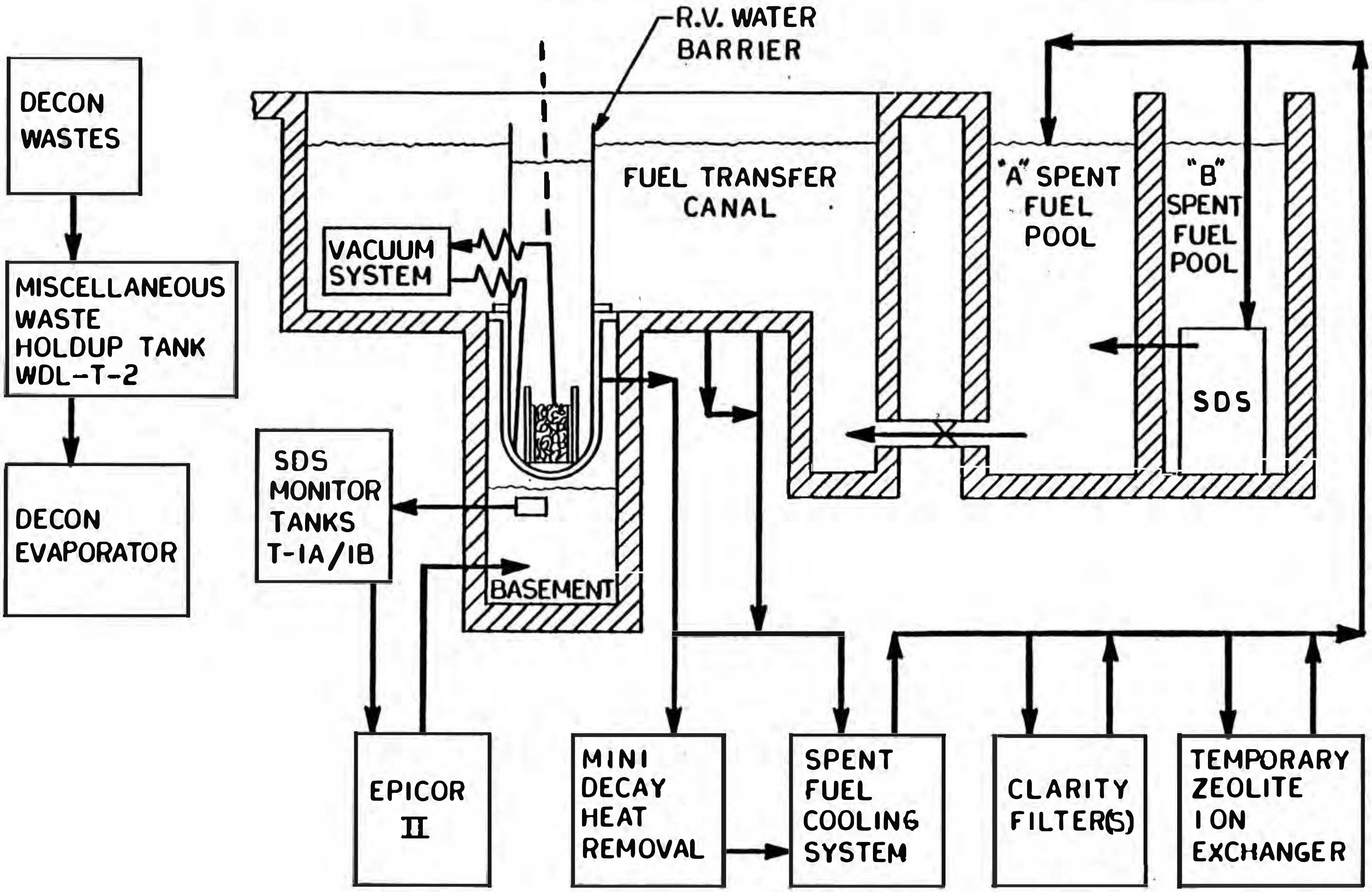
solutions. It can collect and process water from most areas of the plant. It is well shielded and designed for this service. Its only disadvantage is that it utilizes the TMI-1 radwaste evaporator.

A new dedicated radwaste evaporator is necessary to maintain the unit 1/unit 2 separation. An HPD crystallizer was purchased by GPU in 1979, immediately following the accident. This crystallizer is apparently still at the site (Reference 12) and could be used for this purpose.

A temporary solidification system would be needed to process the evaporator bottoms.

FIGURE - 1

DEFUELING/WATER CLEANUP SYSTEM SCHEMATIC



Attachment a

Attachment b

APPENDIX 1

DEFUELING/WATER CLEANUP SYSTEM DESIGN CRITERIA

Defueling System

Purpose:	The defueling system shall be used to remove all fuel debris which can be fluidized hydraulically from the reactor vessel.
Piping Size:	Piping inside diameter shall be 2.5-3 times the largest dimension of fuel debris to be routinely handled. The return line to the reactor vessel shall be sized to keep the fluid velocities as low as practical.
Piping Material:	Piping system shall be fabricated from abrasion-resistant, hard rubber hose.
Flow Rate:	Flow rate shall be adequate to establish a fluid velocity in the slurry piping two times larger than the fuel debris settling velocity.
Debris Collection Canister:	Drop-out canisters shall be utilized for gross debris removal. Cyclone separator canisters shall be utilized for high weight percent solids slurries with uniform particle size (~1-500 microns). High efficiency filters shall be utilized for low weight percent solids slurries with uniform particle size (~1-500 microns) All canisters shall be compatible with the fuel shipping containers which must utilize the existing fuel transfer mechanisms. The K effective of any arrangement of debris collection canisters in a fuel-shipping container shall be less than .95.

All canisters shall be provided with differential pressure instrumentation and a thimble location for a radiation monitor.

All canisters shall be provided with ball check quick disconnect fittings compatible with remote operation.

All canister materials of construction shall be compatible with the water chemistry in the fuel pool and in the reactor vessel, and with the DOE imposed requirements for taking possession of the fuel.

All containers shall have dewatering capability.

All canisters shall be no more than four feet in overall length.

All drop-out canisters shall provide:

- o a drop-out plenum with velocities low enough to assure removal of all fuel debris larger than 1000 microns.
- o a back-washable filter element in the 100-500 micron range.
- o a maximum operating differential pressure at design flow of less than 10 psid.

All cyclone separator canisters shall provide:

- o cyclones with involute feeds
- o cyclones with a 95% removal rate for fuel debris down to 10 microns.
- o cyclones with a maximum differential pressure drop at design flow less than 10 psid.
- o a cyclone feed manifold that is resistant to clogging.

All filters shall provide:

- o filter media with a high surface area.
- o filter media compatible with high radiation doses.
- o a filtering capability of 95% of all particles down to 1 micron in size.

Centrifugal Pump: The centrifugal pump shall be a self-priming pump compatible with motive fluid pressure and flow requirements of the defueling eductor.

Pump materials shall be compatible with the water chemistry of the reactor coolant system.

Pump shall be able to withstand an integrated dose of 1.0E+7 rads over its service life.

Eductor: Eductor shall be able to generate the required system design flow rate at a suction lift of 25 feet.

Eductor shall be able to develop a minimum discharge head of 25 feet at the system design flow rate while generating the maximum design suction lift.

Eductor shall be constructed of materials compatible with the water chemistry of the reactor coolant system.

Eductor shall be able to withstand the water hammer forces resulting from dead-heading the eductor discharge during operation.

Eductor shall be capable of passing an intact fuel pellet through the diffuser portion of the eductor body.

Backwash: The defueling system shall be capable of clearing its suction path by backflushing.

Attachment b

Water Cleanup System

Purpose: The water cleanup system shall be provided to remove fine debris and dissolved radionuclides from the shield water. It will also be able to cleanup the spent fuel pool "A" and the fuel transfer canal.

Installed Systems Used:

- o Decay Heat Removal (DH)
- o Alternated Decay Heat Removal (ADH)
- o Mini-Decay Heat Removal (MDHR)
- o Spent Fuel Cooling System (SF)
- o Submerged Demineralizer System (SDS)

Modifications Required:

- o Install new temporary zeolite ion exchanger to replace the Spent Fuel Demineralizer (SF-K-1)
- o Modify Spent Fuel Filters (SF-F-1A/1B) handling area to be compatible with higher source terms
- o Make tie-in from spent fuel pool return piping to inlet of SDS.

Temporary Zeolite Ion Exchanger

Design:

Flow Rate	200 gpm
Pressure Rating	100 psi
Volume	~90 ft^3

Purpose: To remove radionuclides from process water whenever there is a change in concentration.

Reactor Building Basement Water Cleanup System

Purpose: Maintain the concentration of cesium in the reactor building basement shield water below 1mCi/ml.

Remove ~25 curies of cesium 137 from the basement per day to affect decontamination of the basement.

Systems to be Used:

- o Submerged Demineralizer System (SDS)
- o Epicor II

Modifications Required:

- o Reroute discharge of SDS-P-2 to SDS monitor tanks

Attachment b

- o Tie SDS monitor tanks' discharge into Epicor II piping
- o Tie discharge of Epicor II into monitor tanks

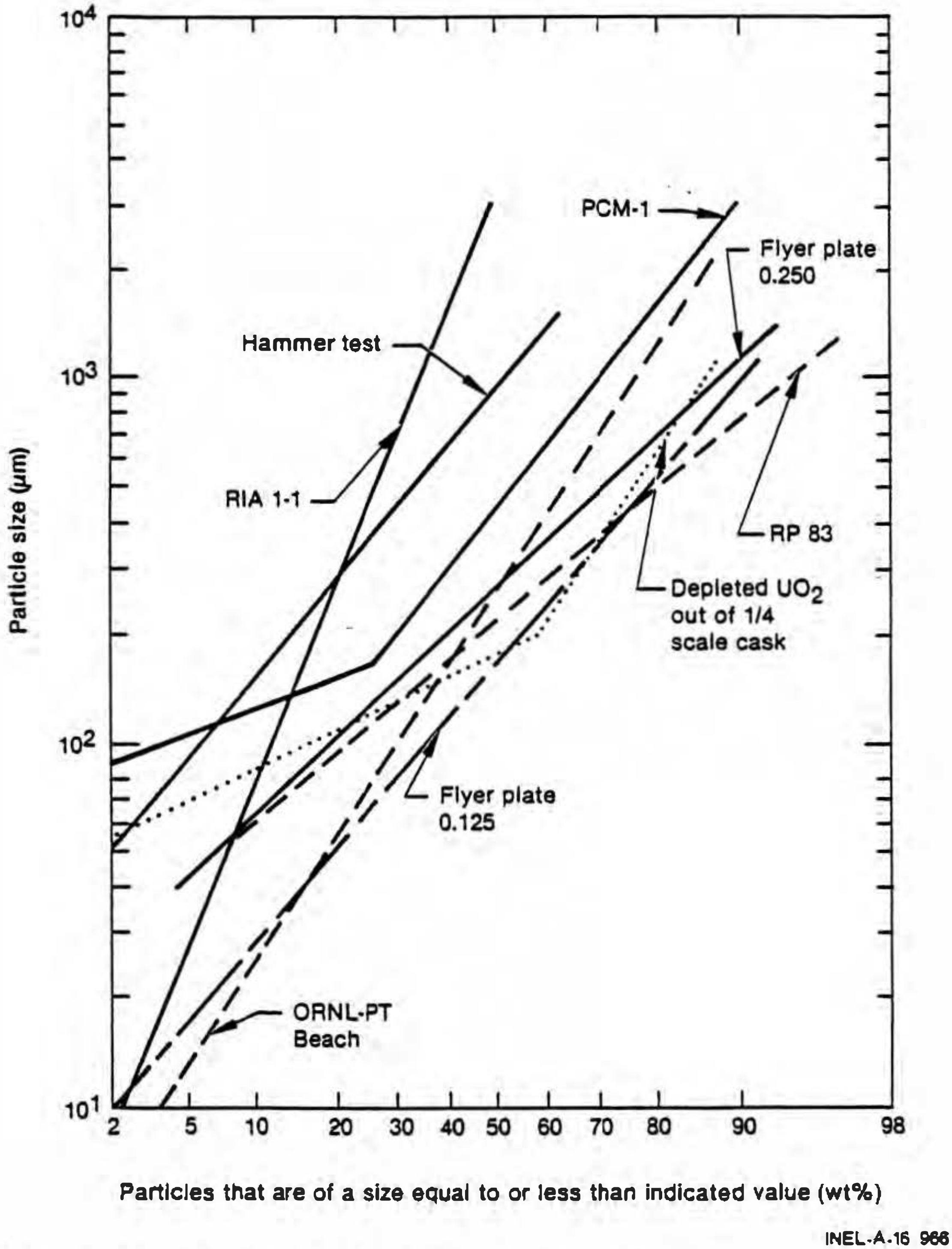


Figure 2. Particle Distribution (Ref. 2)

Attachment d

TABLE 1

UO_2 PARTICLE TERMINAL VELOCITY
(REF. 8)

Particle Diameter (in)	Terminal Velocity (ft/sec)	Required Pump Flow Rate (gpm) (@ 2x Term. Velocity)					
		D=.75"	D=1.0"	D=1.25"	D=1.50"	D=1.75"	D=2.0"
.001	.3	.83	1.5	2.3	3.3	4.5	5.9
.005	.7	1.9	3.4	5.4	7.7	10.5	13.7
.010	.95	2.6	4.7	7.3	10.5	14.3	18.6
.05	2.0	5.5	9.8	15.3	22.1	30.0	39.2
.100	3.0	8.3	14.7	23.0	33.1	45.0	58.8
.150	3.7	10.2	18.0	28.3	40.8	55.5	72.5
.200	4.2	11.6	20.6	32.2	46.3	63.0	82.3
.250	4.7	13.0	23.0	36.0	51.8	70.5	92.1
.300	5.2	14.3	25.5	40.0	57.3	78.0	101.9
.350	5.6	-	27.5	42.9	61.8	84.1	109.8
.363	5.7	-	27.9	43.6	62.9	85.6	111.7
.4	6.0	-	29.4	46.0	66.2	90.1	117.6
Intact Pellet	5.6	-	-	-	-	84.1	109.78

FIGURE-3
DEFUELING ARRANGEMENTS

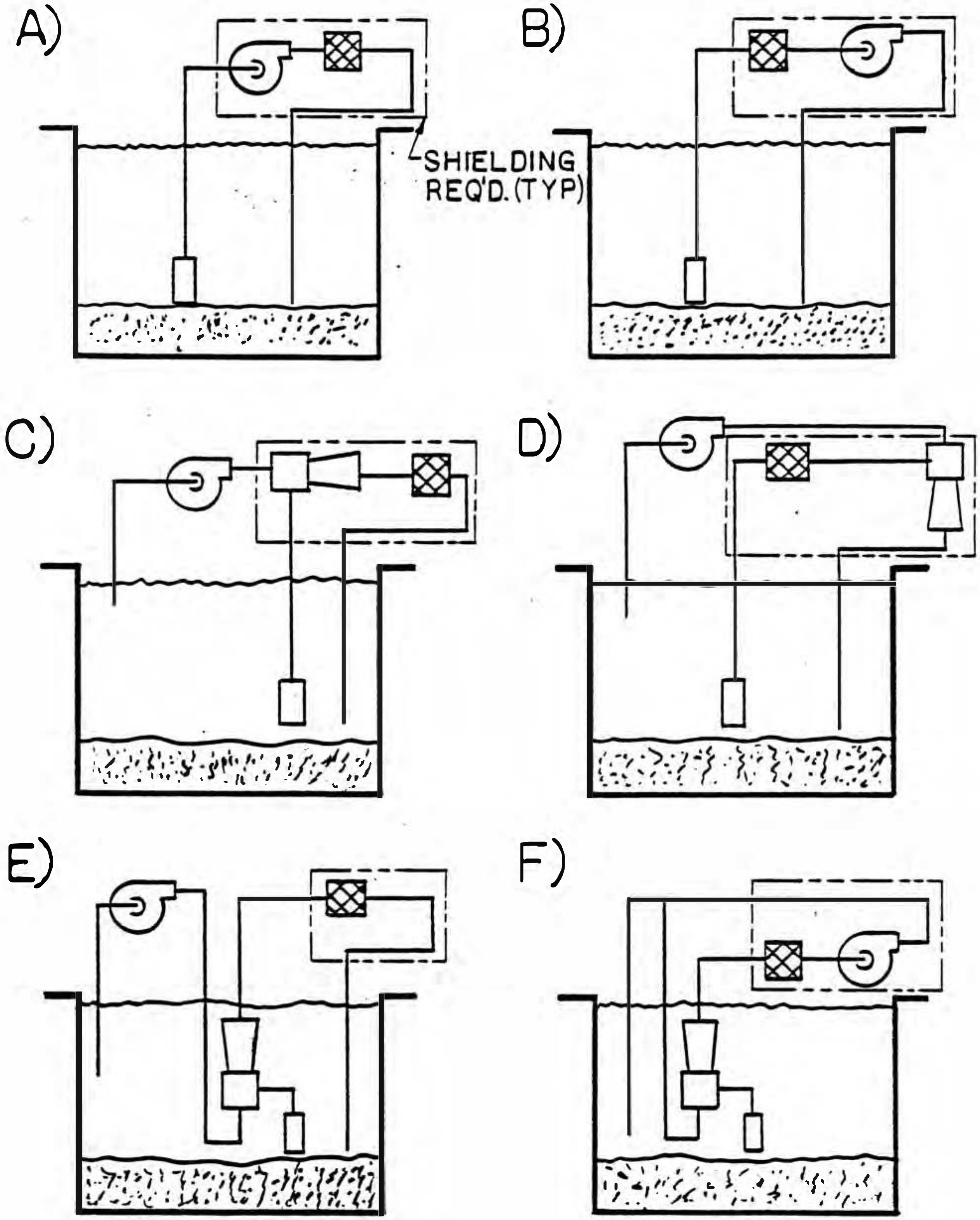


Figure 4 - Typical Solids Handling Eductors

Type 224 Eductors are used in pumping out wells, pits, tanks or sumps where there is an accumulation of sand, mud, or other material not easily handled by the standard eductor. They are ideal for handling the heavy sludge residue from refining operations. A typical application of a Type 224 Eductor is shown in the application section.

These eductors have an open suction and are designed to be submerged in the material being handled. The pressure liquid, passing through the nozzle, produces a high velocity jet which entrains the sludge or mud. This mixture is then discharged through a vertical pipe or hose. For performance information, see accompanying Performance Data.

Similar units which use steam as the motive power are described in Bulletin 2A under "Type 225 Syphons."

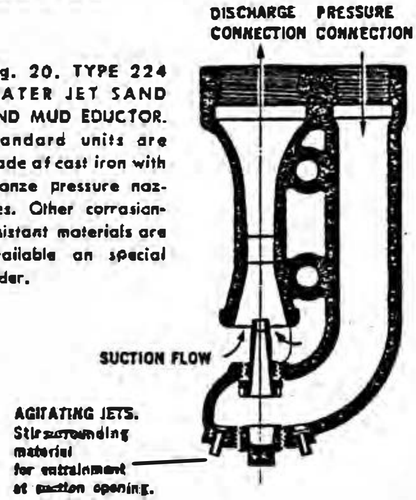


Fig. 20. TYPE 224 WATER JET SAND AND MUD EDUCTOR. Standard units are made of cast iron with bronze pressure nozzles. Other corrosion-resistant materials are available on special order.

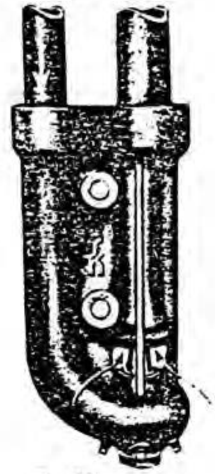
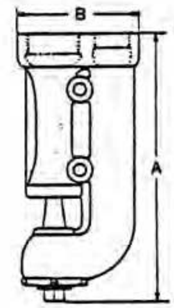


Fig. 21. TYPE 224 EDUCTOR

TABLE 7. SIZES and DIMENSIONS, TYPE 224 EDUCTORS

Size in Inches	Connections in Inches		Wgt. in Lbs.	Dimensions in Inches	
	Disch.	Pressure		A	B
1½	1½	1	8	9⅜	4⅜
2½	2½	2	42	16¾	7⅞
3	3	2½	87	21⅛	10¼
4	4	3	130	25½	11
5*	5	4	....	30¼	17⅜
6*	6	4	....	35 5/16	18

*Flanged Connections.



Type 235 Eductors are designed to handle solids and semi-solids. They operate at highest efficiency in large sizes and at low discharge heads. Because these eductors have high air handling capacities, they are particularly well suited for priming large pumps such as dredging pumps which frequently encounter air pockets. A typical application is shown in the application section.

Nozzles on the periphery of the throat introduce the pressure water. The pressure water creates a vacuum which draws in and entrains the material being handled and all flow discharges through the discharge connection. All suction flow is in a straight line through the eductor. For performance information, see accompanying Performance Data.

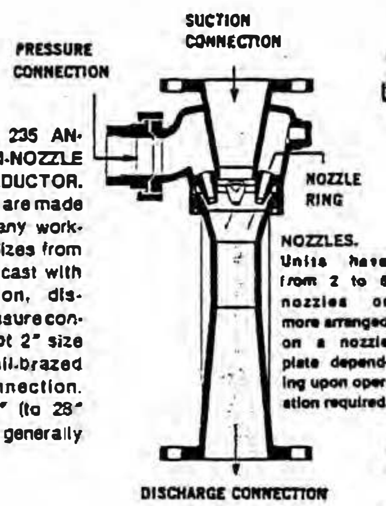


Fig. 22. TYPE 235 ANNULAR MULTI-NOZZLE WATER JET EDUCTOR. These eductors are made to order from any workable material. Sizes from 1½" to 6" are cast with flanged suction, discharge and pressure connections, except 2" size which has sil-brazed pressure connection. Sizes above 6" (to 28" and up) are generally fabricated.

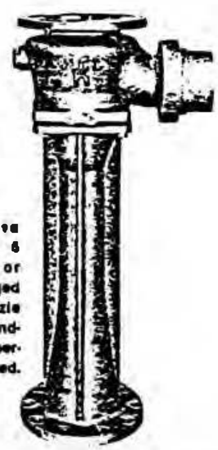


Fig. 23. TYPE 235 EDUCTOR (2" size with sil-brazed connection).

TABLE 8. SIZES and DIMENSIONS of ANNULAR MULTI-NOZZLE WATER JET EDUCTORS, TYPE 235

Size in Inches	Connections in Inches		Wgt. in Lbs.	Dimensions in Inches		
	Suction Disch.	Pressure		A	B	C
1½	1½	1	16	2⅞	8 7/16	3⅜
2	2	1¼	22	3⅛	11⅞	3¾
2½	2½	1½	27	3¼	12 5/16	4
4	4	2½	65	4⅛	18 11/16	5⅛
5	5	3	100	4⅜	24 7/16	6
6	6	4	150	5¼	30	7¼

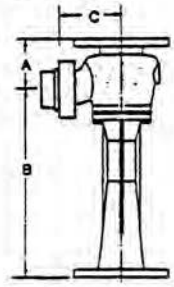
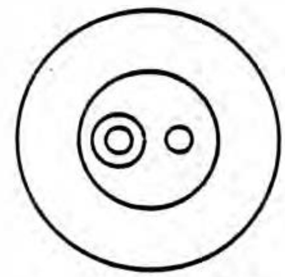
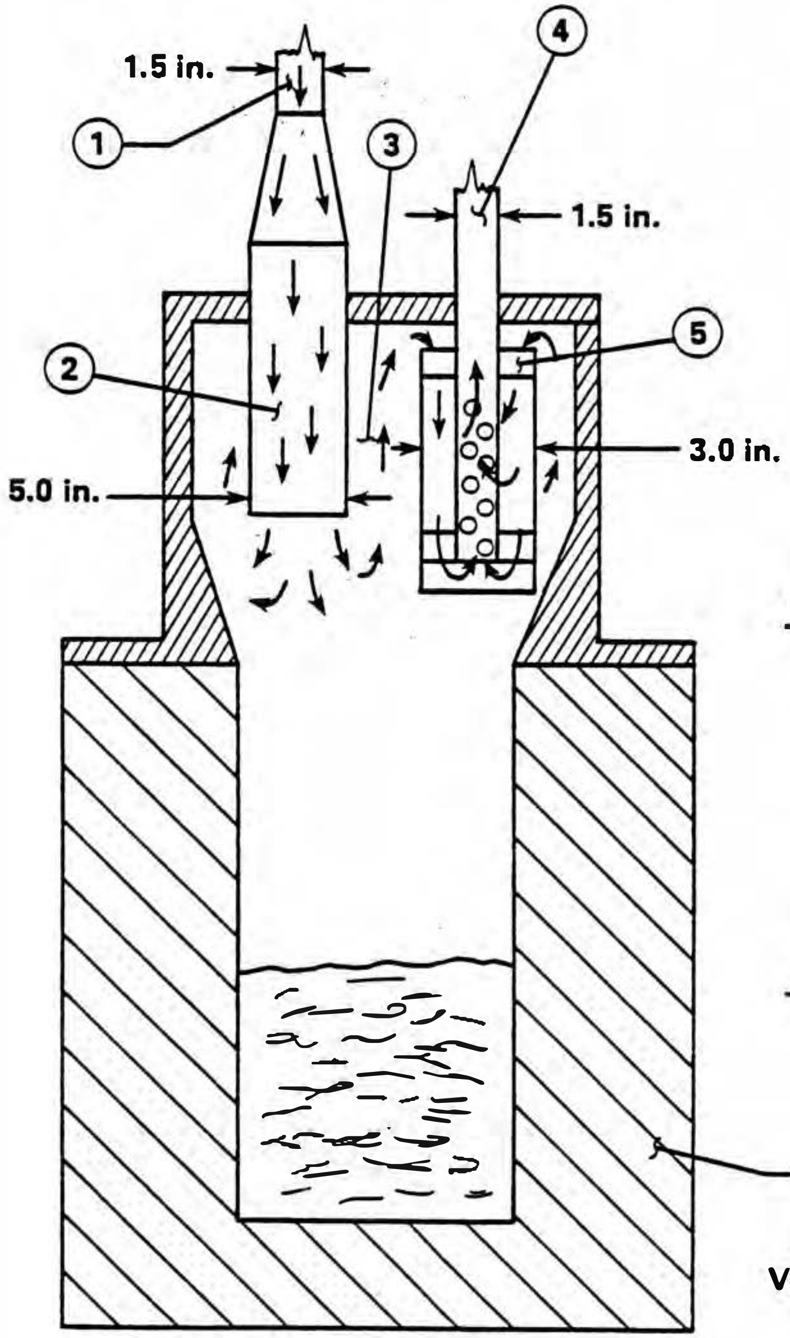


Figure 5

PARTICLE SETTLING CANISTER





FLOW LOCATION	FLOW VELOCITY (ft/sec)
1	12.7
2	0.8
3	0.5
4	12.7
5	3.2

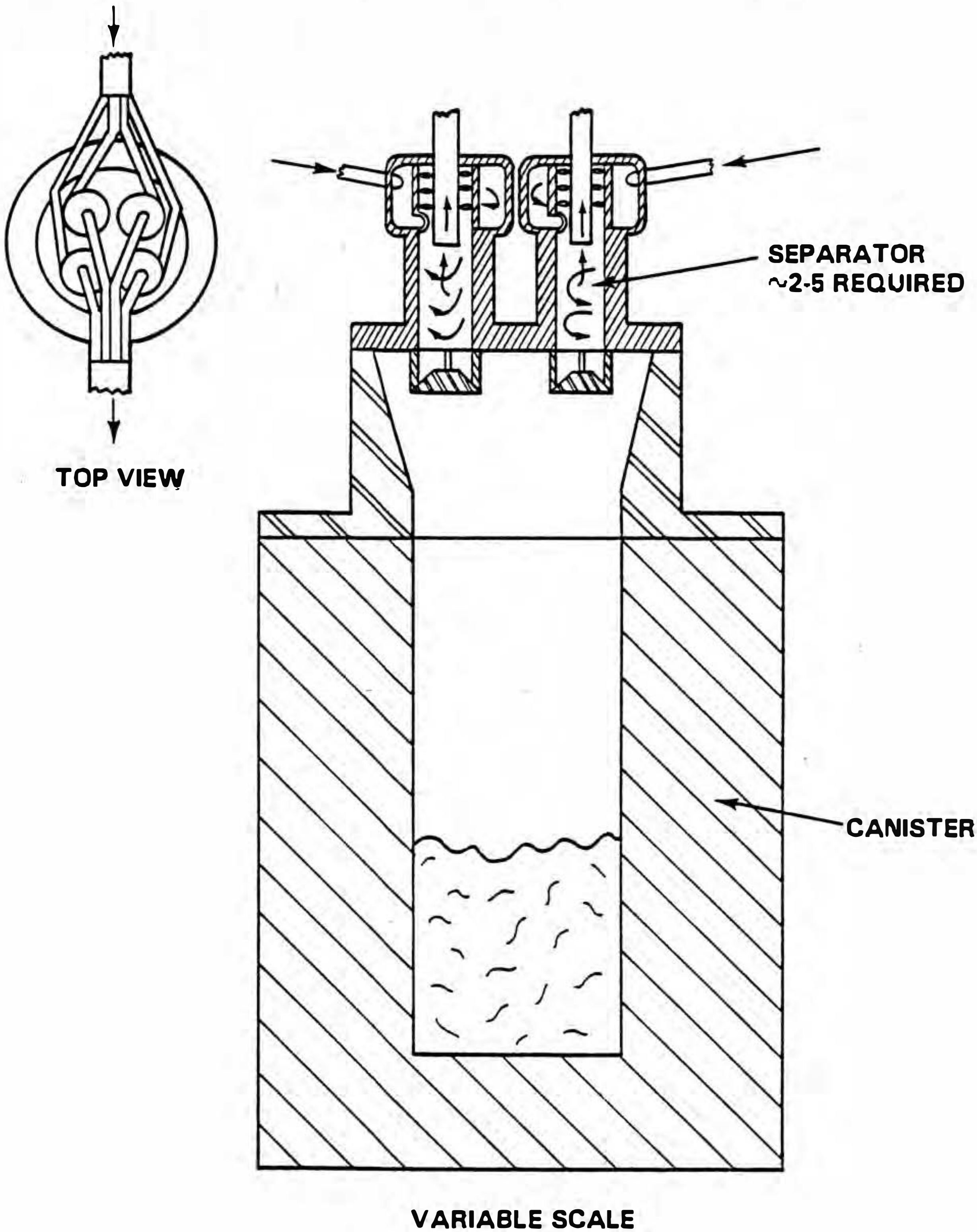
CANISTER

VARIABLE SCALE

HEDL 8302-210.3

Figure 6

CYCLONE SEPARATING CANISTER



VARIABLE SCALE

TABLE 2

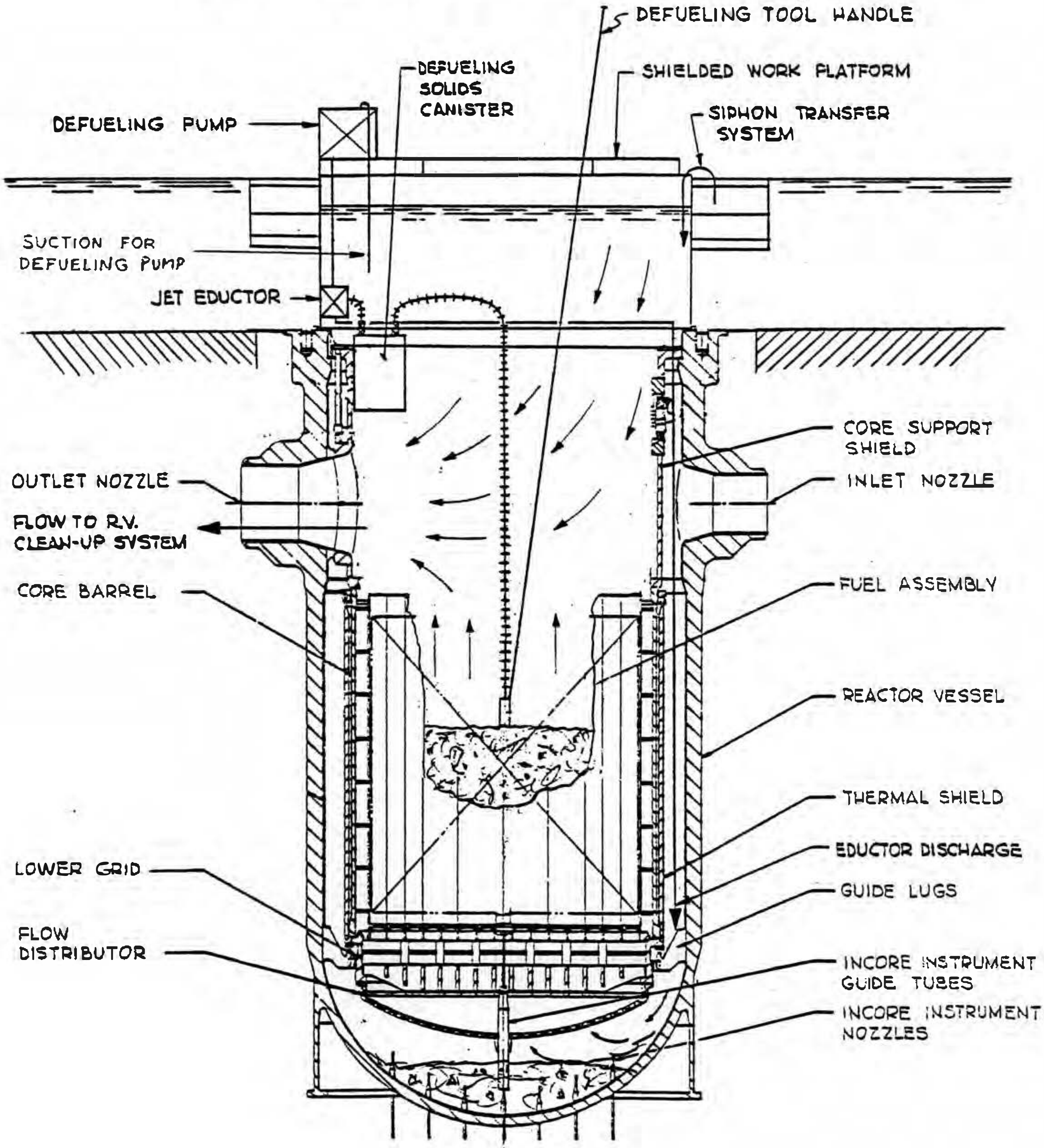
REACTOR VESSEL CLEANUP RATES
(REF. 8)

Contaminated Volume	Gallons	Lbs. of Solids	Initial PPM	*Req'd Time (Hrs) to Reach IPPM @ 100 GPM	@ 200 GPM	@ 400 GPM
Reactor Vessel (RV)	27,842	3,200	14,240	44	22	11
		320	1,424	34	17	9
		32	142	23	12	6
RV & Internal Indexing Fixture (IIF)	35,083	3,200	11,301	55	27	14
		320	1,130	41	21	10
		32	113	28	14	7
RV & Collars Up To Refueling Canal	55,604	3,200	7,130	82	41	21
		320	713	61	30	15
		32	71	40	20	10
GPU Nuclear Volume	78,000	3,200	5,082	111	56	28
		320	508	81	41	21
		32	51	45	23	11
RV Above Nozzles & IIF	17,501	3,200	22,653	29	15	7
		320	2,265	23	11	6
		32	227	16	8	4
RV Above Nozzles & IIF & Collars Up To Refueling Canal	45,263	3,200	8,759	69	34	17
		320	876	51	26	13
		32	88	26	13	7

$$* \ \text{time} = -\ln\left(\frac{\text{IPPM}}{\text{Initial PPM}}\right) \times \left(\frac{\text{Contaminated Vol.}}{\text{GPM}}\right) \quad \underline{\text{or}} \quad \frac{\text{PPM @ } t}{\text{PPM Initial}} = e^{-\frac{(\text{GPM})}{(\text{gal})} t}$$

Attachment j

FIGURE 7
DEFUELING SYSTEM ARRANGEMENT



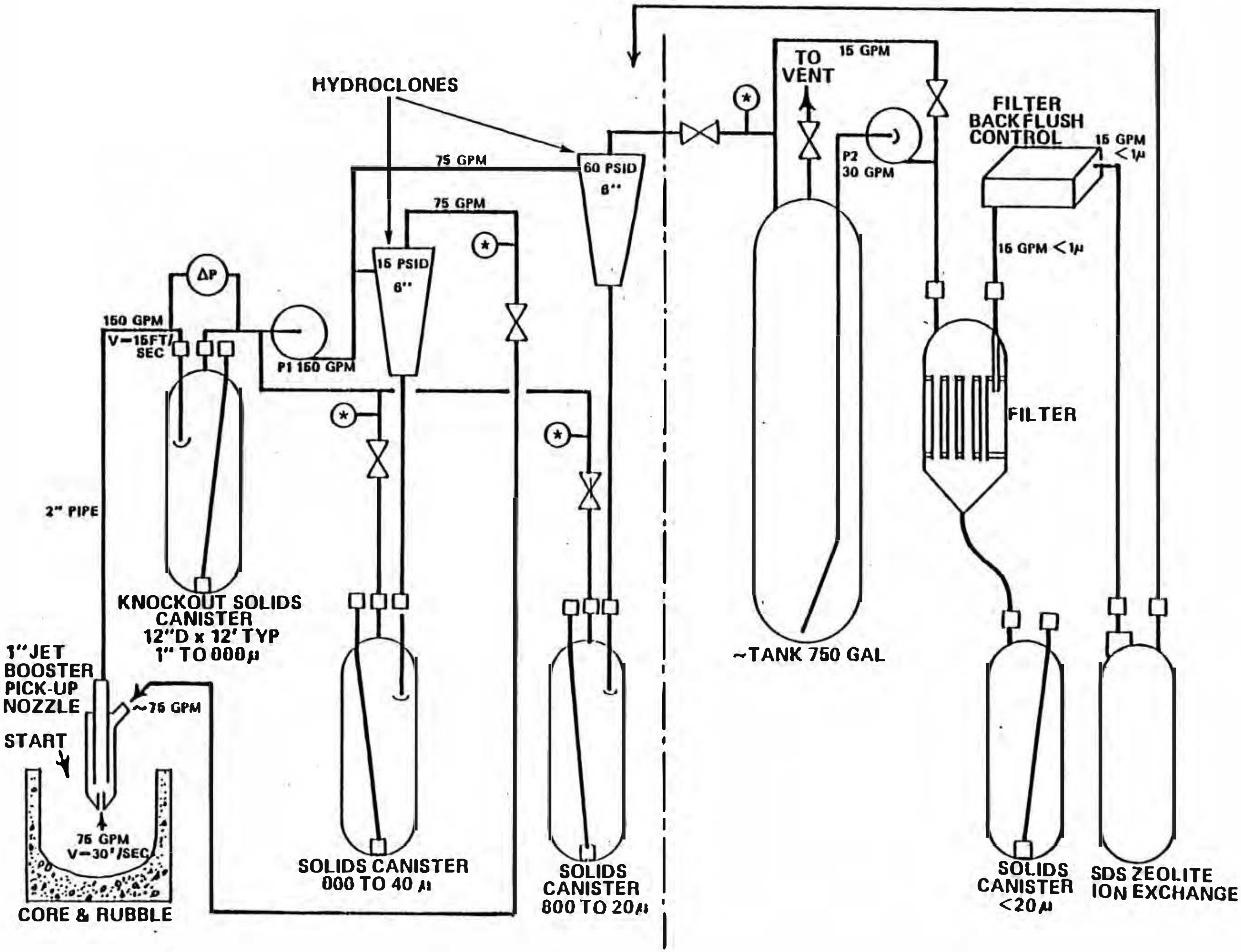


Figure 8 - Work Group Defueling System (Ref. 10)

Attachment K

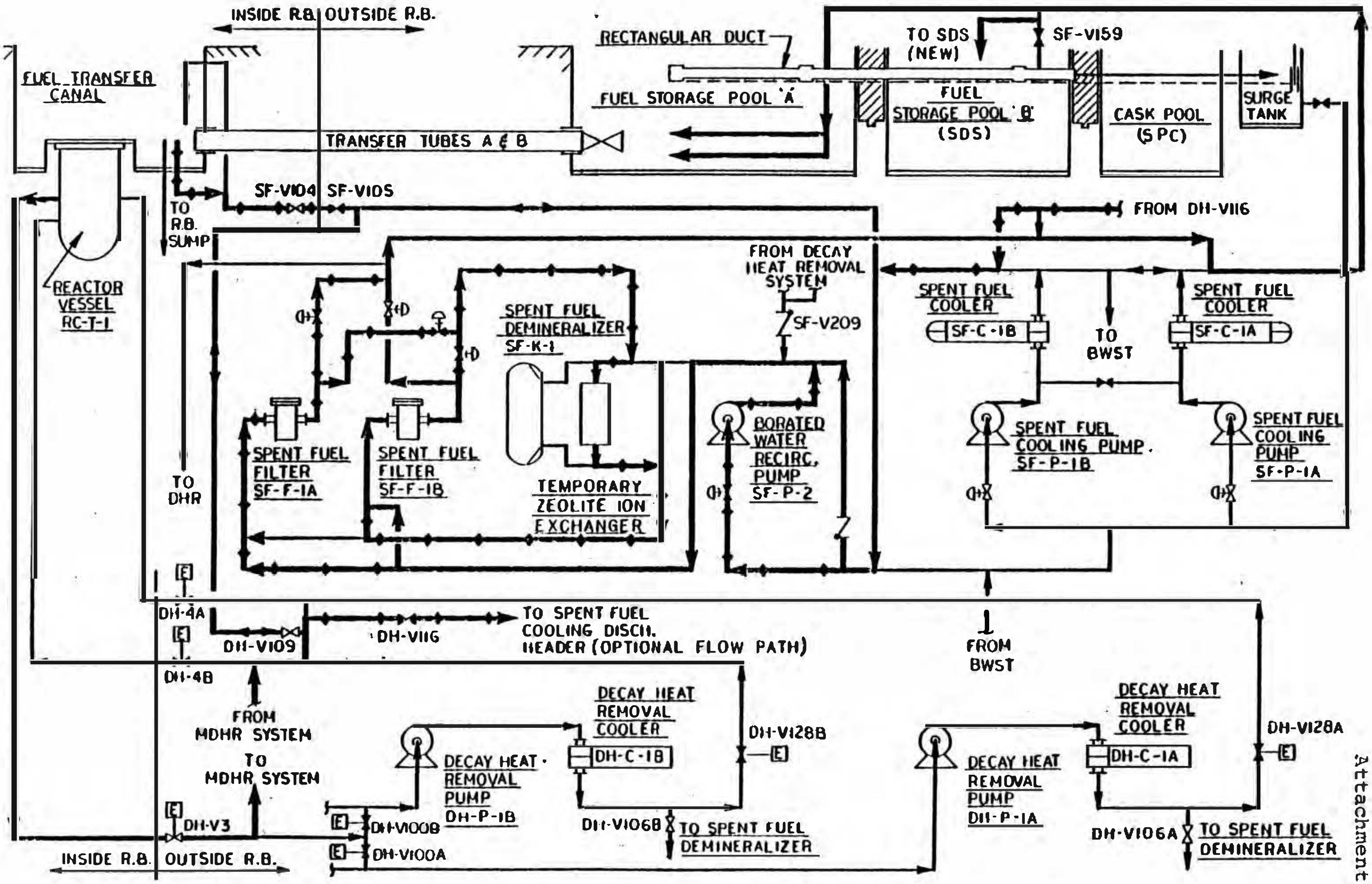


FIGURE 9 - CLEANUP SYSTEM FLOW DIAGRAM

LEGEND

CLEANUP FLOW PATH

OPTIONAL PATH

Attachment 1

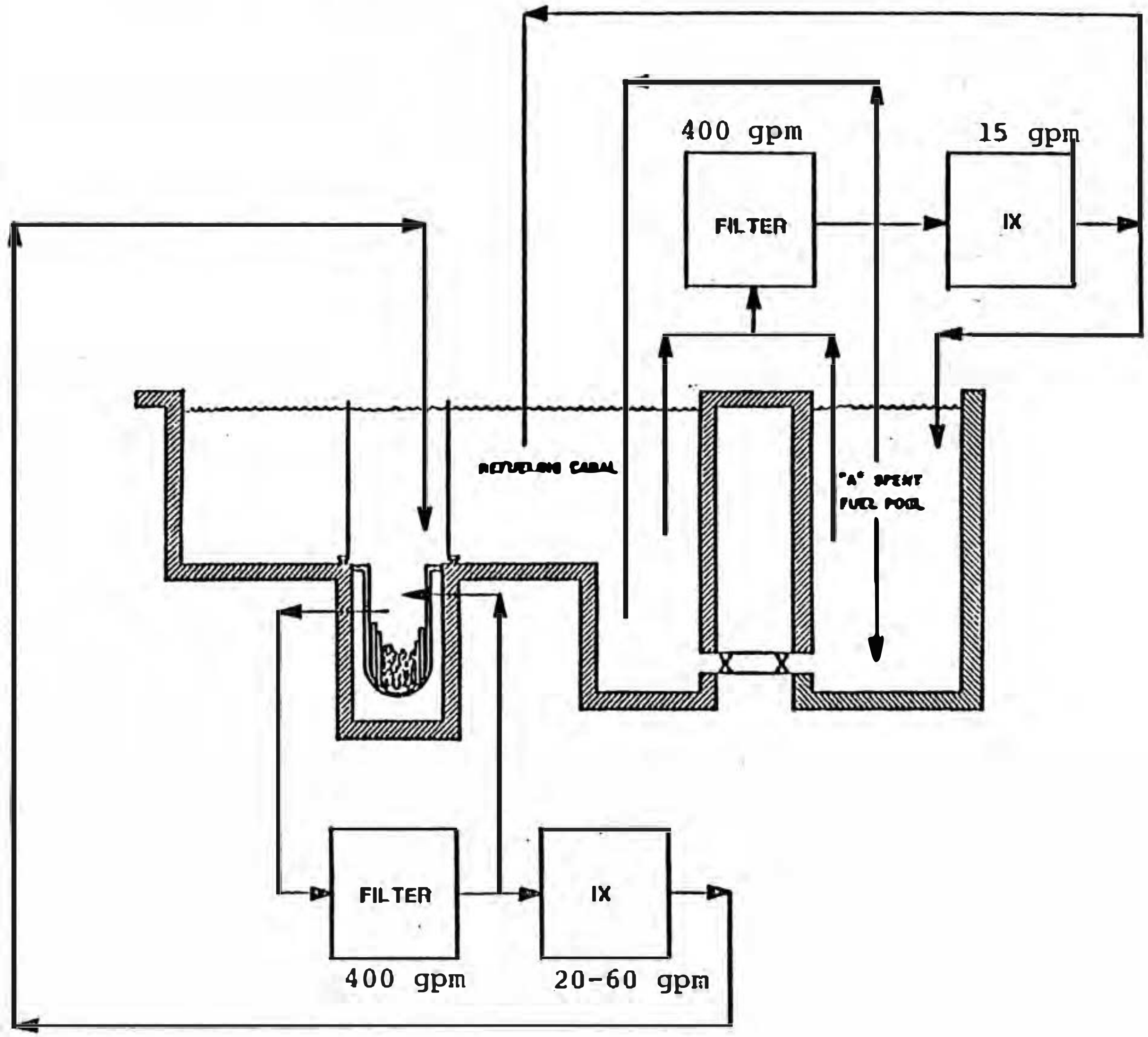
Attachment m

TABLE 3

WATER CLEANUP SYSTEM VALVE LINE UP
(Ref. 6)

VALVE TAG NO.	OPEN = O SHUT = S	VALVE TAG NO.	O S	VALVE TAG NO.	O S
DH-V1	O	MDH-V20	S	SF-V113	O
DH-V2	O	MDH-V12A	O	SF-V20B	O
DH-V3	O	MDH-V15	O	SF-117	S
ADH-V01	O	MDH-V16	S	SF-V116A	O
ADH-V03	S	MDH-V18	O	SF-V115A	S
MDH-V1	O	MDH-V19	O	SF-V120A	S
MDH-V2	O	ADH-07A	O	SF-V121A	O
MDH-V3B	O	DH-108A	O	SF-V125	O
MDH-V28	S	DH-109	O	SF-V161	O
MDH-V30	O	DH-110	S	SF-V150	S
MDH-V35	O	DH-4B	S	SF-V158	S
MDH-V36	O	SF-V155	O	SF-V159	S
MDH-V29	O	SF-V106	O	SF-V240	S
MDH-V32	S	SF-V105	S	SF-V122	S
MDH-V5	O	SF-V115B	O	SF-V214	S
MDH-V6A	O	SF-V207	S	SF-118A	S
MDH-V6B	S	SF-V120B	O	SF-119A	S
MDH-V7A	O	SF-V121B	S	SF-116B	S
				SF-V112	S

Figure 10 - GPUN Cleanup System

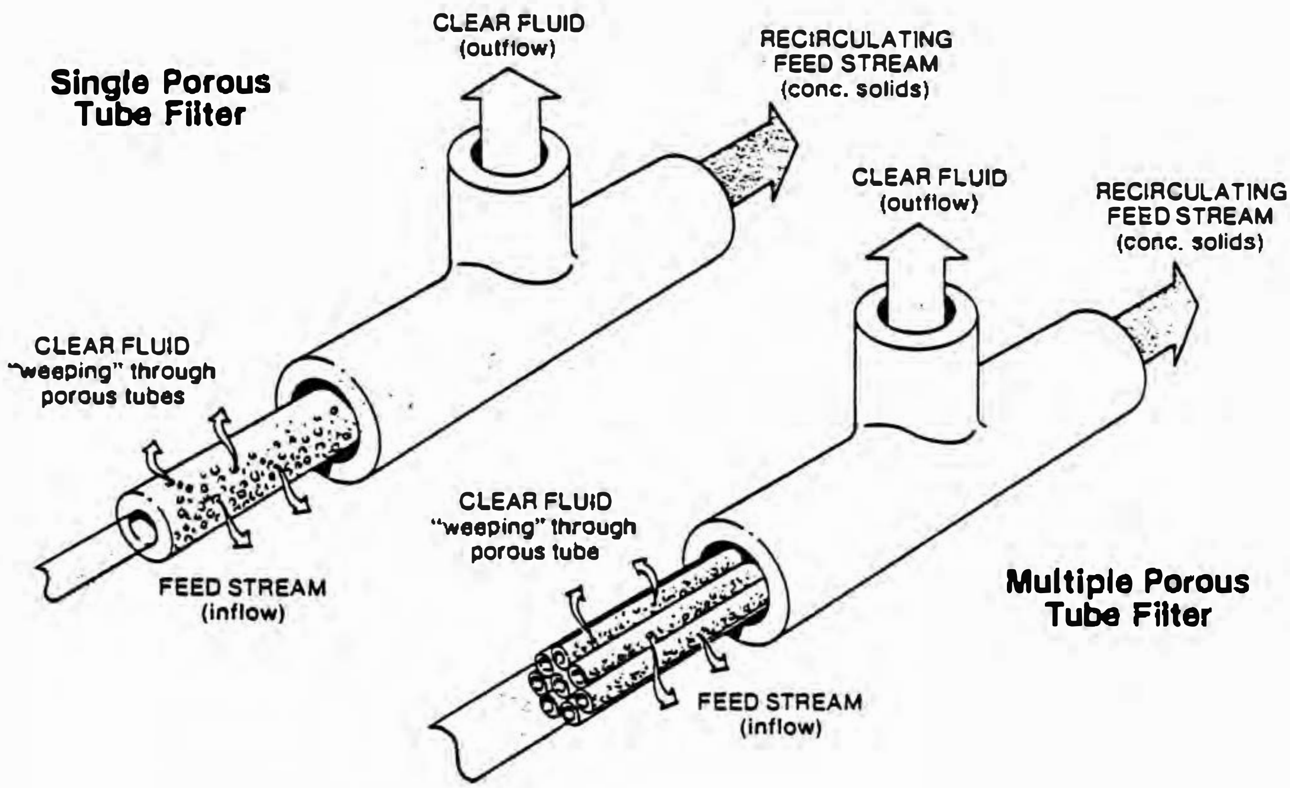


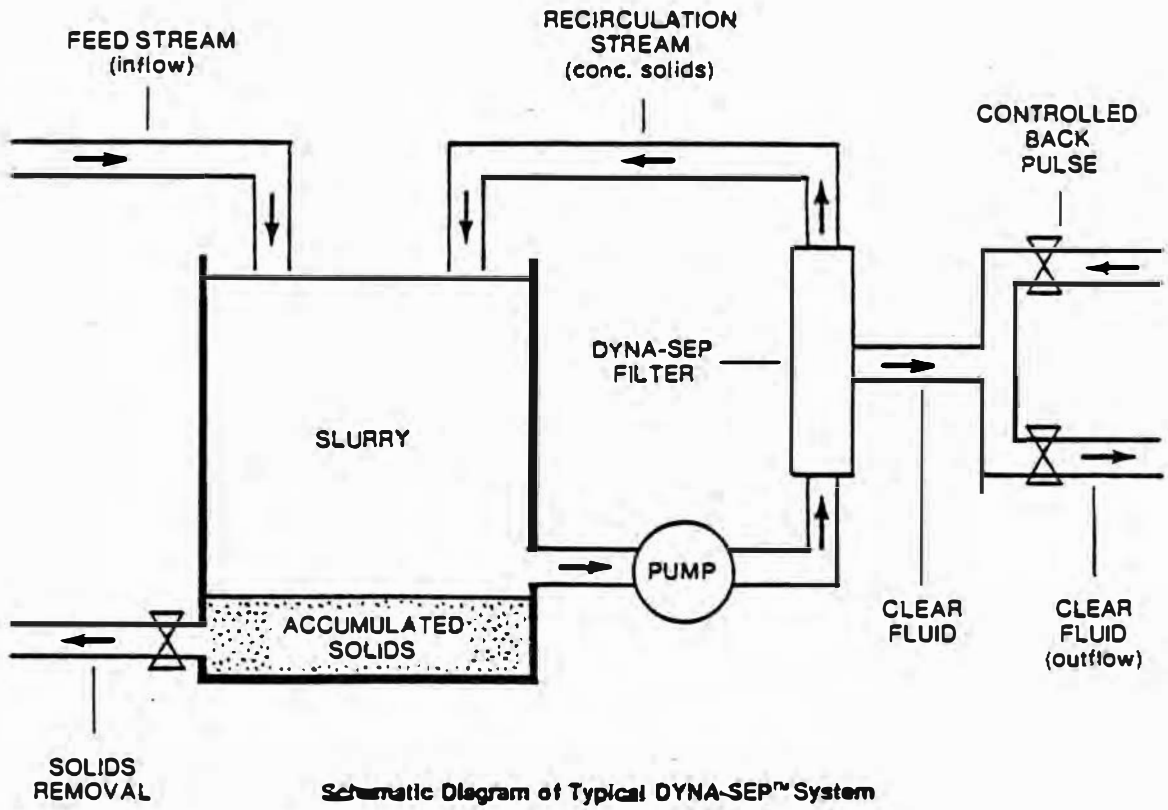
(Ref. 7)

Attachment n

Figure 11 - GPUN Cleanup Filter

Attachment O





Schematic Diagram of Typical DYNA-SEP™ System

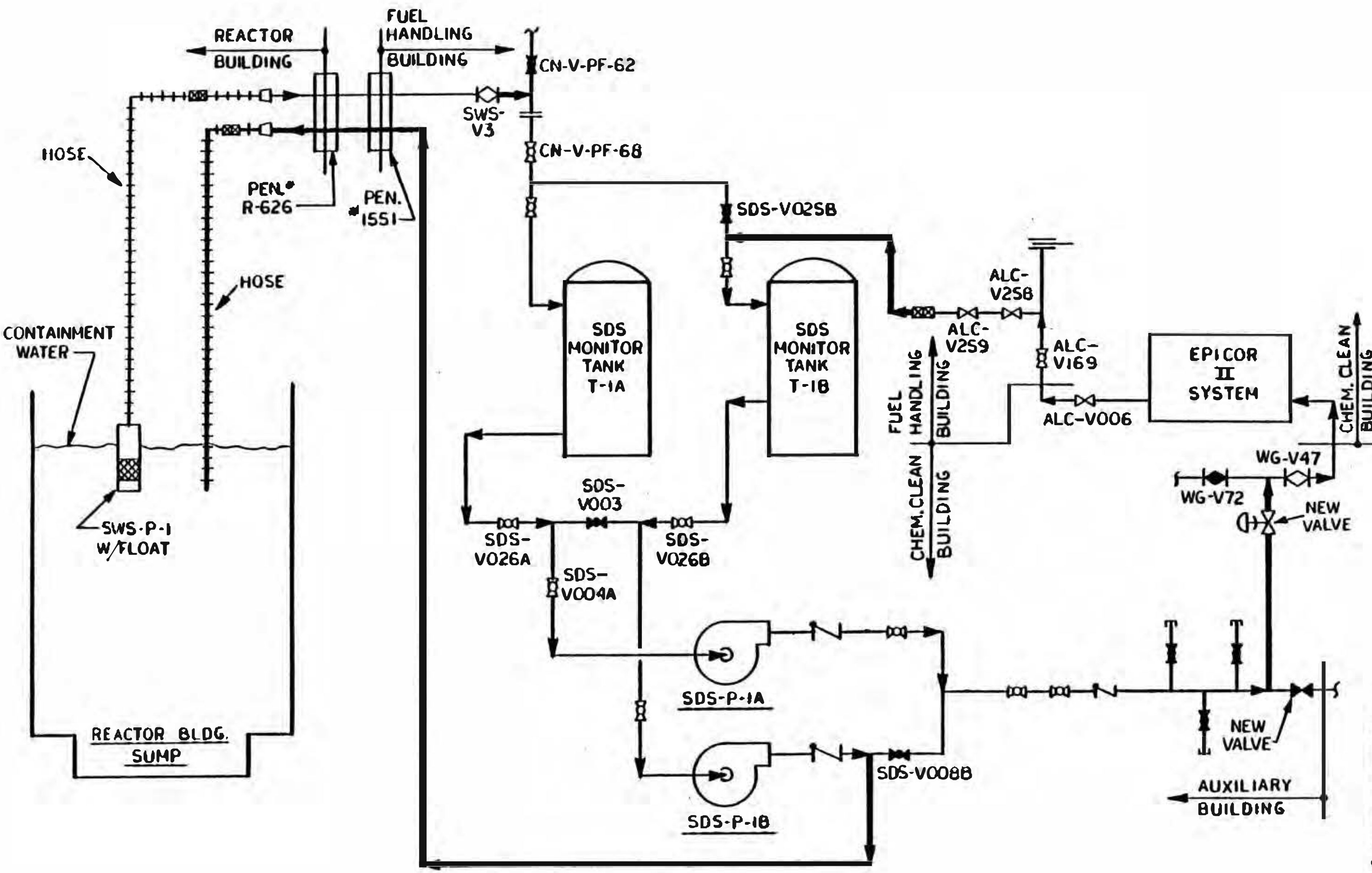


Figure 12 - Epicor II Flow Diagram

Attachment P

APPENDIX IX-1, Core Examination Program

TABLE 3. TMI-2 CORE EXAMINATION RECOMMENDATIONS

Reactor Component or Sample[a]	Examination Priority[b]	Safety Issue to be Addressed[c]	Principal Examination Techniques	Data Obtained	Principal Data Uses	Comments
Leadscrew Guide Sleeves from the Reactor Vessel Dome	Moderate	Fission Product R, T, & D (b, c, d, f)	Gamma scan, radiochemistry, surface analysis, metallography, SEM/microprobe, Acid-Base leaching	Fission product plateout, metal temperatures, extent of oxidation	Fission product transport codes, source term determination, core internals temperature estimates	3 guide sleeves required from center, midradius, and periphery. This exam will complement measurements made by EPRI for reactor vessel head requalification.
Plenum Cover Debris	Low	Fission Product R, T, & D (a, b, e) Coolability/Damage Processes (a, e)	Chemical analysis, SEM/microprobe, radiochemistry, particle size surface analysis, gamma scan	Debris composition and particle size, fission product content of debris	Core debris relocation models, fission product transport codes, source term determination	1 sample required if there is significant accumulation observed by CCTV inspection
Plenum Cover Specimens	Low	Fission Product R, T, & D (b, d, f)	Radiochemistry, surface analysis, metallography, gamma scan, SEM, Acid-Base leaching	Fission product plateout, metal temperatures	Fission product transport codes, core temperature codes, plenum temperature estimates	~5 specimens required (punchings or cuttings)--locations based on visual examination
Radiation Mapping of Plenum	Moderate	Fission Product R, T, & D (f)	Gamma scan	Isotopic radiation levels as a function of position in the plenum	Fission product transport codes	This exam needed to guide later samplin of plenum. Recommend be performed after head removal by lifting and gamma scanning all leadscrews. Could be done by alternate technique after plenum removal
Control Rod Leadscrews	V. High	Fission Product R, T, & D (b, c, d, f)	Gamma scan, surface analysis, radiochemistry, metallography, SEM/STEM/microprobe	Fission product plateout, component temperatures, extent of oxidation	Fission product transport codes, plenum temperature estimates, core exit steam temperature calculations	The three leadscrews removed in 1982 are probably adequate
Split-tube Sections	V. High	Fission Product R, T, & D (b, c, d, f)	Surface analysis, gammascan, radiochemistry, metallography, SEM/microprobe, Acid-Base leaching	Fission product plateout, component temperatures, extent of oxidation	Fission product transport codes, plenum temperature estimates, source term determination	Short sections cut from selected split-tubes at all radial locations and 2-3 axial locations. These specimens will provide an early thorough mapping of fission product plateout on plenum
Control Rod Guide Tube Assemblies	High	Fission Product R, T, & D (b, c, d, f)	Photo-visual, gamma scan, surface analysis, radiochemistry, SEM, metallography, Acid-Base leaching	Fission product plateout, peak temperature estimates, extent of component deformation and melting	Fission product transport codes, source term determination, core and plenum temperature codes	3 complete assemblies from center, midradius, and peripheral positions should be obtained at the time of plenum disassembly
Control Rod Spiders	High	Fission Product R, T, & D (d, f)	Photo-visual, surface analysis, metallography, radiochemistry, SEM, gamma scan, Acid-Base leaching	Component temperatures, extent of oxidation, fission product plateout	Core temperature codes, H_2 generation estimates, fission product transport codes	~6 spiders concentrated at the core center and midradius positions
Fuel Assembly End Fittings	High	Fission Product R, T, & D (d, f)	Photo-visual, surface analysis, metallography, radiochemistry, SEM, gamma scan, Acid-Base leaching	Fission product plateout, component temperatures, extent of oxidation	Fission product transport codes, source term determination, core exit steam temperature calculations, H_2 generation estimates	~6 end fittings from same locations as spiders
Intact Fuel Assemblies	High	Fission Product R, T, & D (a, b, d) Coolability/Damage Processes (c, f) Recriticality (a) Appendix K (a, b)	Photo-visual, gamma scan, metallography, radiochemistry, surface analysis, SEM, acid-base leaching, burnup analysis	Cladding, fuel and control rod temperatures, oxide distribution, cladding deformation, flow blockages, fission products retained in the fuel, fission product plateout	Appendix K issues (zircaloy oxidation and embrittlement, hydrogen generation, and zircaloy cladding ballooning and flow blockage); fission product retention; fission product plateout, source term determination	3 assemblies req'd, 1 from core periphery
Damaged Fuel Assemblies	Critical	Fission Product R, T, & D (a, b, c, d, e) Coolability/Damage Processes (a, c, d, e, f) Containment Integrity (a, b) Recriticality (a) Appendix K (a, b)	Photo-visual, neutron tomography, metallography, chemical analysis, SEM, particle size, radiochemistry, microprobe, gamma scan, surface analysis	Cladding, fuel and control rod temperatures, extent of oxidation, extent of eutectic melting and fuel liquefaction, flow blockage, fuel rod fragmentation and relocation, UO_2 oxidation, fission product release from fuel, relocation of control materials	Core temperature codes, core debris relocation models, H_2 generation estimates, fission product transport codes, source term calculations, recriticality analysis, flow blockage models, liquid material movement models	3 assemblies req'd, 1 w/o control rods, 1 w/control rods, 1 w/burnable poison rods

a. Listed by physical location within the reactor vessel, proceeding from the top of the vessel down.

b. The examination priorities for each type of reactor component or sample are a measure of the impact of the examination data on the combined nuclear safety issues described in Table 3.

c. Lower case letters refer to specific data needs listed in Table 2.

TABLE 3. TMI-2 CORE EXAMINATION RECOMMENDATIONS (Continued)

Reactor Component or Sample	Examination Priority	Safety Issue to be Addressed	Principal Examination Techniques	Data Obtained	Principal Data Uses	Comments
Loose Debris Specimens	Critical	Fission Product R, T, & D (a, b, d, e, f) Coolability/Damage Processes (a, c, d, e, g) Containment Integrity (a, b) Recriticality (a)	Photo-visual, metallography, chemical analysis, SEM, particle size, radiochemistry, microprobe, surface analysis, gamma scan, Acid-Base leaching, density	Fuel and structural material reactions, relocation of core materials, extent of fragmentation, extent of oxidation, retained fission products, nature of debris stratification, peak core temperatures, in-core instrument damage	Debris bed coolability models, fission product transport codes, core debris relocation models, source term determination, H_2 generation estimates, in-core instrument survivability analysis, recriticality analysis	~40 specimens of mtls. such as fuel and cladding pieces, control mtls, spacer grids, fuel rod springs liquefied mtl., in-core instruments, structural mtls Efforts will be made to acquire some specimens in a manner which preserves stratification so that the sequence of damage events can be reconstructed
Crust Debris Specimens	Critical	Fission Product R, T, & D (a, b, d, e, f) Coolability/Damage Processes (a, b, c, d, e) Containment Integrity (a, b) Recriticality (a)	Photo-visual, metallography, chemical analysis, SEM, radiochemistry, microprobe, gamma scan, surface analysis	Molten material relocation; fuel, control, and structural material reactions; extent of oxidation; retained fission products; peak core temperatures; control material relocation	Core debris relocation models, fission product transport codes, debris bed coolability models, H_2 generation estimates, source term determination, recriticality analysis	~10 specimens of core debris in which significant quantities of once-molten material is present
Fuel Stub Assemblies	V. High	Fission Product R, T, & D (a, b, d, e,) Coolability/Damage Processes (a, c, d, e) Recriticality (a)	Photo-visual, neutron tomography, metallography, radiochemistry, SEM and microprobe, gamma scan, surface analysis	Microstrucure of damage transition zones, extent of oxidation, retained fission products, relocation of core materials	Liquid level boildown models, fission product codes, core debris relocation models, H_2 generation estimates, recriticality analysis	3 stub assemblies, 1 from the small number of gadolinia-bearing experimental assemblies in the cor-
Loose Debris from Lower Vessel	Moderate	Fission Product R, T, & D (e) Coolability/Damage Processes (a, b) Containment Integrity (c, d) Recriticality (a)	Photo-visual, metallography, chemical analysis, SEM, particle size, radiochemistry, microprobe, gamma scan surface analysis, Acid-Base leaching, density	Estimate of total quantity, particle size distribution, extent of once-molten debris, fission product content	Core debris relocation models, fission product transport codes, vessel breach models, recriticality analysis, debris bed coolability	~5 samples of debris on the various horizontal surfaces below the core, particularly the bottom head. Sample selection should be based on CCTV inspection
Core Former Wall	Moderate	Fission Product R, T, & D (d, f)	Metallography, radiochemistry, surface analysis, gamma scan SEM, Acid-Base leaching	Fission product plateout, peak metal temperature, extent of oxidation	Fission product transport codes, radiative heat loss models	~2 punchings or cuttings adjacent to an intact assembly
Filter Debris	High	Fission Product R, T, & D (a, e, g) Coolability/Damage Processes (a) Recriticality (a)	Photo-visual, metallography, chemical analysis, SEM, particle size, radiochemistry, microprobe, physical properties	Retained fission products, particle size distributions, fuel control material, and structural material reactions, relocation of core materials	Fission product transport codes, fuel fragmentation models, core relocation models, source term determination, recriticality analysis	Refers to analysis of samples from Makeup and Purification System filters which plugged with core debris during the accident
Debris Specimens from Balance of Reactor Coolant System	High	Fission Product R, T, & D (a, b, c, e, g) Coolability/Damage Processes (a) Recriticality (a)	Photo-visual, metallography, chemical analysis, SEM, particle size, radiochemistry microprobe, physical properties	Retained fission products, particle size distribution, core materials reactions, relocation of core materials	Fission product transport codes, core relocation models, criticality analysis, source term determination, mass balance determination	Debris in filters, tanks, pipes, containment sump, etc. should be quantified. Partial overlap with EPRI responsibilities for requalification
General Documentation of Large-Scale Condition of the Reactor Vessel and Core	Critical	Fission Product R, T, & D (a, e, f) Coolability/Damage Processes (a, b, d, e, f) Containment Integrity (a, b, c, d) Recriticality (a)	Photo-visual, closed circuit TV, core topography mapping	Core damage symmetry, core void size, total volume and mass of debris, stub assembly elevations, extent of liquefaction, transition zone configurations, major coolant channels	Core relocation models, recriticality analysis, core coolability models, mass balance determination, molten material relocation models, nature of debris stratification	This data will assist in planning and conducting all other examinations