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September 14, 1984

9/20/84

Distribution

TRANSMITTAL OF NINTH TAAG REPORT - Hmb-212-84

Dear Sirs:

Enclosed herewith is the ninth report of the Technical Assistance and Advisory Group (TAAG). This report covers the TAAG activities during the period from April 1, 1984 to August 1, 1984.

Very truly yours,

H. M. Burton

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

mrr

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NINTH REPORT

OF THE

Technical Assistance and Advisory Group (TAAG)

Three Mile Island Unit 2 (TMI-2)

For the Period April 1, 1984 to August 1, 1984

<u>TAAG MEMBERS</u>

W. H. Hamilton, Chairman
R. Brodsky
N. M. Cole
E. A. Evans
T. A. Hendrickson/C. W. Hess
A. P. Malinauskas/E. D. Collins

<u>Participating</u>

F. A. Ross, DOE
H. M. Burton, EG&G
R. Weller, NRC
A. Roberts, EPRI

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SECTION 1.0

INTRODUCTION

In a letter dated April 25, 1984, Mr. B.K. Kanga, Director, Three Mile Island Unit 2 (TMI-2), requested that the Technical Assistance and Advisory Group (TAAG) address the following matters for the period ending August 1, 1984:

1. Review the results of demonstration tests of the Defueling Water Cleanup System filters. Evaluate the system (canal water and reactor vessel) for possible simplification.

2. Keep current on defueling plans and provide technical comments as designs are finalized.

3. Review the proposals being made by the GEND Technical Evaluation Group (Core Evaluation for Fission Product Release) for data acquisition and make recommendations regarding the utility of the data and impact on the defueling work.

4. Review the Safety Evaluation Reports (in preparation) for plenum removal and defueling.

5. Assist in resolution of the questions regarding the contribution of sources in the 282', 305', and 347' levels to general radiation levels. Individuals suggested by TAAG have been assigned to this work and will be working closely with the GPUN staff.

6. With regard to locating fuel in the RCS, review the GPUN planning work for detecting the presence of fuel, including instruments to be used and methods of placing the instruments near the piping systems. As necessary, obtain the services of a qualified expert to participate in this review.

7. Evaluate as to whether defueling operations should continue in the event of leaks from the reactor vessel.

8. Evaluate the technical pros and cons for removal of the equipment hatch.

9. Prepare a report on fission product behavior in fuel during defueling.

10. Provide information on a contingency boronometer including vendor, cost, operating experience and why it is better than the existing instrument.

11. Review the design and testing of filters for the canal water clean-up system.

1.1 SCOPE AND APPROACH

This report responds to these work items. One section of the report addresses each of the work items. The recommendations are summarized in Section 2. In the last section of this report the GPU Nuclear responses to the Eighth TAAG Report recommendations are tabulated.

SECTION 2.0

RECOMMENDATIONS

2.1 OWCS FILTER TEST EVALUATION

- o If testing of Pall Trinity production filter elements supports the very favorable laboratory scale results, this type of filter should be incorporated into the DWCS design.

- o If the performance of Pall Trinity production filter elements is unacceptable, the use of sintered metal tubes as filter media should include knockout canisters upstream of these filters; this may reduce the frequency of back bumping the filters.

- o Because the use of deep bed filters is a proven technology, efforts should be made to retain their use as a contingency in the event that unforeseen problems develop with the sintered metal filters.

- o The selection of "dry" defueling, and the attendant use of a shielded platform atop the Internals Indexing Fixture (IIF), have modified the original design criteria for the DWCS. The system should be re-evaluated in light of these modifications.

- o The DWCS design should also reflect considerations of (1) means to prevent overloading of the filter canisters, and (2) protection to prevent sudden rupture of a sintered metal filter. The design should also accommodate the sudden rupture of a loaded filter.

2.2 DEFUELING PLANS

- Westinghouse should be asked to develop three layout sketches to demonstrate the work platform configuration for vacuuming, long handled tools alone, and automated/remote tools.

- Modify the work platform arrangements to better suit defueling with long handled tools (see Section 4 for specifics) and to reduce crowding in the platform center.

- Provide for eventually lowering the work platform onto the reactor vessel flange.

- Westinghouse should be asked to illustrate the procedure for carousel removal.

- Enclose tool lifting cables to prevent airborne contamination problems.

- Provide a contingency to add a shielded transfer boot extending down from the work platform. This should be done because radiation streaming may be a problem for work continuation while canisters are being removed.

2.3 TEG RECOMMENDATIONS FOR DATA ACQUISITION

The only controversial issue related to the Technical Evaluation Group (TEG) recommendations for core data acquisition is core boring. Since DOE is proceeding on this matter, TAAG has discontinued efforts on this subject.

2.4 REVIEW SAFETY EVALUATION REPORTS (SERs) FOR PLENUM REMOVAL AND DEFUELING

TAAG recommends that the SER for plenum removal be issued as a single report and that maximum use be made of previously issued SERs.

2.5 DOSE SOURCE IDENTIFICATION

TAAG has no recommendations on this subject during this period.

2.6 LOCATING FUEL IN THE RCS

TAAG has identified alternate methods of placing radiation instruments underneath the vessel and in the vicinity of the letdown coolers. The use of either of these paths should be considered in data acquisition planning.

2.7 DEFUELING WITH LEAKS

TAAG recommends that defueling can continue with leaks so long as the water level in the reactor vessel is maintained. It is noted that such operations are allowed at all commercial PWR power plants.

2.8 EQUIPMENT HATCH REMOVAL

TAAG recommends that the containment equipment hatch be removed to perform a job or set of jobs and then be replaced. Large items should be staged outside the reactor building as much as possible to reduce the number of times and the duration of time that the equipment hatch is removed. Special measures should be taken to reduce the environmental risks associated with opening the equipment hatch and these measures should be evoked only while the equipment hatch is open. A draft SER has been sent to GPUN under separate cover.

2.9 FISSION PRODUCT BEHAVIOR IN FUEL DURING DEFUELING

A method for estimating fission product release during defueling has been reported in TPO/TMI-133. After estimating fission products currently distributed within the fuel, releases were evaluated 1) as the remainder of the fuel is "crushed" and new surfaces are created during defueling, and 2) by subsequent leaching. The results of TPO/TMI-133 may be used for planning defueling water cleanup system operations.

2.10 BORONOMETER

An evaluation should be conducted of the feasibility of using conductivity meters to monitor boron concentration.

SECTION 3.0

DWCS FILTER TEST EVALUATION

TAAG was asked to review and to evaluate the defueling water cleanup system (DWCS) filter demonstration testing, and to recommend ways of simplifying the DWCS. This item has been carried over for the past two reporting periods largely due to operational difficulties at the B&W test facility. This delay has affected the timeliness of this evaluation; the DWCS design is finalized and procurement efforts are ongoing. Thus, any simplification, or change in design strategy at this point, regardless of its merits, could delay the schedule and might adversely affect the start of bulk defueling. However, the need to have the DWCS emplaced and fully operational in order to begin defueling should be re-examined in light of the "dry" defueling scheme selected.

The testing program comprises full-length, single-element tests with simulated reactor coolant. The tests used two different types of filter elements. The first type was a sintered metal tube, with the flow from the inside out. The second type was a pleated-sintered metal element, with the flow from the outside in. Two vendors were selected for the first type of element: Mott and ATI. The second type of filter was provided by Pall Trinity.

The sintered metal tubes have the disadvantage of requiring periodic back bumps to remove flow-blocking deposits. These back bumps are sudden pressure pulses applied to the discharge of the filter to dislodge particles trapped in the porous structure of the sintered metal. Each back bump is followed by a quiescent period to permit the solids to settle down the tube to the collection plenum at the bottom of the assembly. The hardware required for the back bump system is elaborate.

If the solids loading is high, the frequency of back bumping approaches the duration of a back bump and the system would be constantly cycled as filters are removed from the process stream in rapid succession to be cleared. If this occurred, the wear and tear on the equipment could result in significant maintenance outages during the operating life of the DWCS.

The second type of filter does not need to be back bumped. This has operational as well as capital cost advantages over the first type of filter. It can be used in a simpler system with fewer valves, less pipe, and no pressure accumulators. It is a straightforward pump/filter system with easier operation and less wear and tear on the components. In addition, the filter element performed better than did the sintered metal tube type. For these reasons, the pleated-sintered metal element has been selected as the leading type of filter to be used for the DWCS. TAAG concurs in this judgment as long as the followup tests with production-quality elements are as promising as the tests run with the prototype element. If this Pall Trinity filter element is selected, the DWCS can be greatly simplified by the deletion of the back bump piping and valves.

If testing of the Pall Trinity production elements indicates that those filters are unacceptable, a knockout canister upstream of the sintered metal tubes may reduce the solids content enough to enable a reasonable duration of time between back bumps.

A generic operational problem with sintered metal filter elements is that they are brittle. Often, the shut off head discharge pressure of the pump is sufficient to break the element, causing the filter to fail. This feature of the design of the DWCS must be addressed if sintered metal filters are to be used. TAAG continues to believe that a backup to the use of sintered metal filters (e.g., deep bed filters) should be actively considered in planning activities.

The final approved defueling plan cuts in half the volume of water to be processed and reduces the water clarity requirements. Hence, the DWCS, which was sized on 78,000 gallons, is at least a factor of two too large. In addition to reducing the volume of water to be processed, the new defueling concept employs a shielded work platform that reduces the serious effects of contaminated water. The elimination of the in-vessel shredder from the defueling concept has also reduced the need for a high throughput system since a major source of particulate generation has been removed. The use of television cameras close to the working end of the tools will reduce the need for water clarity, which also reduces the need for a high throughput system. It seems prudent to re-examine the design of the DWCS in light of changes in the design criteria.

In addition to reducing the need for a high throughput DWCS in the reactor vessel, the new defueling plan radically alters the water processing situation in the fuel transfer canal. The volume of water is reduced from 324,000 gallons to approximately 70,000 gallons. All direct contact of the canal water with the reactor vessel water has been eliminated by multiple barriers.

The only source of contamination in the canal is contaminants adhering to the sides of the debris canisters after transfer from the reactor vessel to the fuel transfer equipment. Hence, the need for the DWCS to clean up the fuel transfer canal is greatly diminished.

Concerns about visibility during the early defueling efforts (i.e., vacuuming) do not seem to be warranted. Hence, the defueling effort will not have need for the DWCS. The solids removal action of the vacuum system and the shielded work platform will be adequate for the removal of the debris bed. Moreover, much of the vacuuming could be performed under conditions of low visibility.

In summary, TAAG believes that the testing program is adequate to identify operational problems of the filter elements. If changes in the water chemistry and the boron concentration are included in future testing and the effects of actual reactor coolant samples are evaluated, the testing program will be acceptable. TAAG believes that the design of the DWCS should be reexamined; TAAG feels that the system may be overdesigned because of recent changes in defueling strategy.

If the Pall Trinity filter element design is successful, the back bump subsystem can be deleted. If the sintered metal tube filter element is chosen, a knockout canister should be placed upstream of the filters to decrease the frequency of back bumps. In either case, final design of the system should not preclude the use of deep bed filters as a contingency.

Loading of the filter canisters apparently will be monitored by noting changes in pressure drop across the filter elements. TAAG was also advised that there is a loading limit for the canisters, although the basis for this limit was unclear. Because estimates of the loading capacity of the filters are undoubtedly conservative, and because pressure drop measurements are only incidentally related to weight capacity, TAAG recommends (1) that the likelihood of overloading the canisters be examined, and (2) if shown to be feasible, a procedure to recover from this situation be developed.

The DWCS design should also accommodate the sudden rupture of a loaded filter.

SECTION 4.0

DEFUELING PLANS

TAAG has continued to review the defueling plans. Comments on some of the early conceptual drawings are included, as Attachment 1, on the following pages.

July 23, 1983

ATTACHMENT 1

Comments on Conceptual Layouts of the TMI-2 Defueling Method Being Developed by Westinghouse

BACKGROUND:

The defueling concept sketches reviewed are entitled:

1. Early and Bulk Defueling System - J. Mino dated 6/15/84

2. Bulk Defueling System - Marchetti dated 6/15/84

This is a general overview of these conceptual layouts. hile it is fully recognized that the concept is evolving and these sketches are conceptual in nature, our observations and comments at this stage may be of help in developing the final version of the TMI-2 defueling concept.

I. General Observations

A. Originally there were to be two different work platforms for use over the reactor vessel:

- One simple static-type work platform was to be particularly directed towards the use of manual long handled defueling tools.

- A second rotating work platform was to be particularly directed toward the use of automated/remote tools; however, it was not to preclude the use of manual tools.

These general directions were set forth in Kintner's memorandum to Kanga dated June 11, 1984 (see items 5 & 6). However, since that time, the effort on the simple static work platform has dropped. The continuing effort design appears to be focused on a rotating work platform that is tailored for automated/remote tooling.

While we understand the original directive, we believe that things have changed in view of dropping the original static platform and we suggest a more balanced approach be taken in the design of the remaining single work platform. Specifically, the platform design should proceed in a manner so that TMI-2 can be effectively manually defueled with long handled tools without being dependent on the automated/remote tooling. In this regard, the concepts shown on the above Westinghouse drawings should be modified to be more useful in the event that a manual long handled tool defueling concept is used. We believe that we can still have a single work platform concept that strikes more of a balance to permit the effective use of the long handled tools while still being able to use the automated/remote tools, but not be totally dependent on them.

B. To provide a means of assuring that the rotating work platform concept finally selected can be effectively used with long handled defueling tools, we would suggest that Westinghouse be requested to develop three layout sketches. Specifically:

1. This first sketch would show how the work platform would be configured for vacuuming loose core debris with long handled tools and no automated/remote tools.

2. A second sketch would show how the proposed work platform would be used to defuel with just long handled defueling tools (i.e., no ROSA or any the other automated tools).

3. The third sketch would show how that same work platform concept could be used with the more automated/remote tools. The third sketch should also show how long handled tools can be used to solve problems that might develop with the automated tools and how the long handled tools still could be used for limited defueling operations in conjunction with automated tools.

II. General Comments

A. The arrangement layout of the carousel, the working slot and the rotatable mast on the work platform (as shown on the sketch identified MINO-6/15/84) results in the following:

1. Makes the structural design and contruction of the work platform more complex than necessary.

2. Makes for unnecessary crowded conditions when loading canisters into the transfer cask. For example, the carousel drive mechanisms, the transfer casks and the rotatable mast are all crowded into the center portion of the work platform. Positioning of the transfer cask in between the carousel drive mechanism and the rotatable mast will be tight and in an area where it will be easy to make mistakes that can damage defueling equipment.

3. Provides a limited size (i.e., small) working slot from which long handled defueling tools can be operated. (Note: This appears to be due to the fact that Westinghouse originally was to develop a work platform tailored around automated/remote equipment and, therefore, this design is not well-suited for long handled tool defueling - see Section I.A above on this issue.)

In view of the above, it is suggested that the work platform arrangement be modified along the following general guidelines:

1. Move the carousel off to one side so that the main structural beams on either side of the working slot can be extended the entire

diameter of the working platform. This will make for a much better and simpler structural design.

2. Make the working slot extend the full diameter of the working platform. (This makes the platform more effectively work for long handled defueling tools.)

3. Locate the transfer cask's transfer point toward the outer end of the working slot, (i.e., toward the ID of the reactor vessel and away from the center as presently positioned).

4. Position the rotatable mast as far from the carousel and the cask transfer position as reasonably practical, but still on the opposite side of the working slot from the carousel.

The separation afforded by the above arrangement should help avoid the crowding around the center area of the work platform and help avoid positioning problems when using the transfer cask. Also, it provides a better arrangement for use of long handled defueling tools. The longer working slot should allow more flexibility for operating the tools and better access to the core area while not encumbering the use of the automated/remote tools.

B. With the conceptual arrangement shown in the above Westinghouse sketches, the working lengths of tooling to reach the top of the lower grid plate are as follows:

- Working water depth: 29'6".
- The distance between the top of the working platform where workers will have to stand and the top of the lower grid plate is 34'.

This arrangement should not present problems with using long handled defueling tools effectively during the early phases of the defueling operations since the depth of the debris will be 5' to 8' less than the depths given above. However, as defueling gets to the lower grid plate region, the depths become as listed above. Working with long handled tools at these depths becomes very marginal and the operation will have to be mostly dependent on automated/remote tools. Further, damage in the region of the lower grid plate and below may be such that long handled tools may be the most effective way to deal with such a situation. Accordingly, it is recommended that the work platform and associated tooling be such that the IIAF and support structures can be removed and the work platform can be lowered down directly onto the reactor vessel flange. This would allow the above listed working depths to be reduced by up to 7' (i.e., to the 22'6" and 27' lengths, respectively). This will help ensure that the defueling concept is not totally dependent on automated/remote tooling for the latter stages of debris and fuel removal. Specifically, this feature will help ensure that long handled tools can be used effectively in the final stages of defueling where we may run into some of the more difficult type operations. (A sketch should be prepared showing how the final work platform concept and its support services can be lowered down on the RV flange. This will help assure that the system really has the capability to be lowered.)

C. It is not clear from these layouts if the carousel is removable in the event that jamming should occur. It is suggested that Westinghouse be requested to show how the carousel, as well as any other "installed" type tooling, would be removed. In this regard, it may be that a smaller carousel might be more appropriate (e.g., be more easily removable, not impact the size of the working slot, etc.)

D. To evaluate defueling concepts and the tooling that is to be used, water depths for shielding should be set so tool-lifting heights can be judged and defined. Specifically, you want to ensure that during loading of canisters with various tools highly radioactive elements are not raised too close to the surface of the water. Basically, workers should be able to open doors in the working slot in the event that automated tool or canister loading operations encounter problems without also having to deal with a significant radiation problem. In essence, the design should not be dependent on shielding in the work platform to handle such situations because the shield door may very well have to be opened to obtain access to solve a problem.

E. Telescoping Tools - We note that there are some telescoping tools that use cables to hoist the retractable masts. It's been our experience that cables coming in and out of water can become a source of airborne activity under some conditions. With the TMI coolant being what it is, bringing cables in and out of water, as well as telescoping tools, may create airborne problems. Therefore, if these types of tools are to be used, they should be enclosed so that they do not create airborne problems.

F. Without a shielded boot extended down from the transfer cask loading position, long handled tool operations through the working slot will have to stop when loaded canisters are being removed. Also, lack of the shielded boot may cause radiation streams at the edge of the work platform when loaded canisters are being raised into the transfer cask. Accordingly, it may be prudent to have the ability to easily add such a transfer boot if radiation streaming or stopping of defueling operation during transfer operations does in fact become a problem.

SECTION 5.0

TEG RECOMMENDATIONS FOR DATA ACQUISITION

TAAG was requested to make recommendations regarding the utility of the data being requested by the GEND Technical Evaluation Group (TEG) and also to assess the impact of the data acquisition on the defueling work. A review of the data being requested was made by TAAG.

The data that were the most debatable were the proposed core bore results. Prior to any further investigation of this matter by TAAG, a decision was made by DOE to proceed with the development of the core boring equipment and to defer until after core debris vacuuming the decision as to whether or not to attempt the boring operation.

With this course of action, TAAG discontinued any further effort on the matter.

SECTION 6.0

REVIEW SAFETY EVALUATION REPORTS (SERs) FOR PLENUM REMOVAL AND DEFUELING

During the period of this review, TAAG was requested to review safety evaluation reports relating to plenum removal and defueling. Those reports are still in a draft state. Brief discussions were held with GPUN personnel on matters relating to these forthcoming reports. Based upon these discussions, TAAG has the following comments:

1. GPUN is planning to issue the SER for plenum removal in three phases. TAAG does not believe it is necessary to issue three reports and recommends issuing one report.

2. TAAG considers that the SERs for the APSR testing, "Quick Look", and head lift combined with the existing in-vessel inspection results can be used to show that the plenum lift operations can be performed safely. Only unique areas such as evaluating the effect of an in-vessel leak of hydraulic fluid should be included in the SER evaluation. Otherwise, the SER should reference the previously approved SERs.

SECTION 7.0

DOSE SOURCE IDENTIFICATION

TAAG was requested to provide assistance to GPU in the ongoing program to investigate and characterize the sources of radiation in the containment building. Paul J. Babel, Burns & Roe, was designated the participant on behalf of TAAG in developing further requirements in the survey and analysis efforts. This effort was ongoing during this report period, and TAAG has had several progress reports during the monthly TAAG meetings. A report summarizing activities to date is scheduled for the next report period.

SECTION 8.0

LOCATING FUEL IN THE RCS

TAAG was requested to review the GPUN work being done to develop instruments and methods to detect the presence of fuel in the RCS but outside the core region. As a followup activity, TAAG was asked to identify any long lead time equipment needs that might be required during the recovery program.

The GPUN planning study describing the Technical Planning Group's approach to this problem is entitled Location and Characterization of Fuel Debris in TMI-2, TPO/TMI-051. Moreover, this approach has been presented to TAAG by the responsible site personnel. TAAG endorses the approach presented and has no comments on the proposed plan.

Due to high dose rates, many areas of interest cannot be included in the current plan. These areas will eventually need to be surveyed to locate and to quantify fuel debris. TAAG does not believe that any new long-lead time radiation instrumentation is required to perform these surveys. The instruments developed for planning study TPO/TMI-051 and other instruments available in the industry should be adequate to perform the surveys required.

A more pressing difficulty is physically placing the detectors near the equipment and piping of interest. TAAG reviewed the layout and design of the plant and has identified pathways to two additional areas of interest.

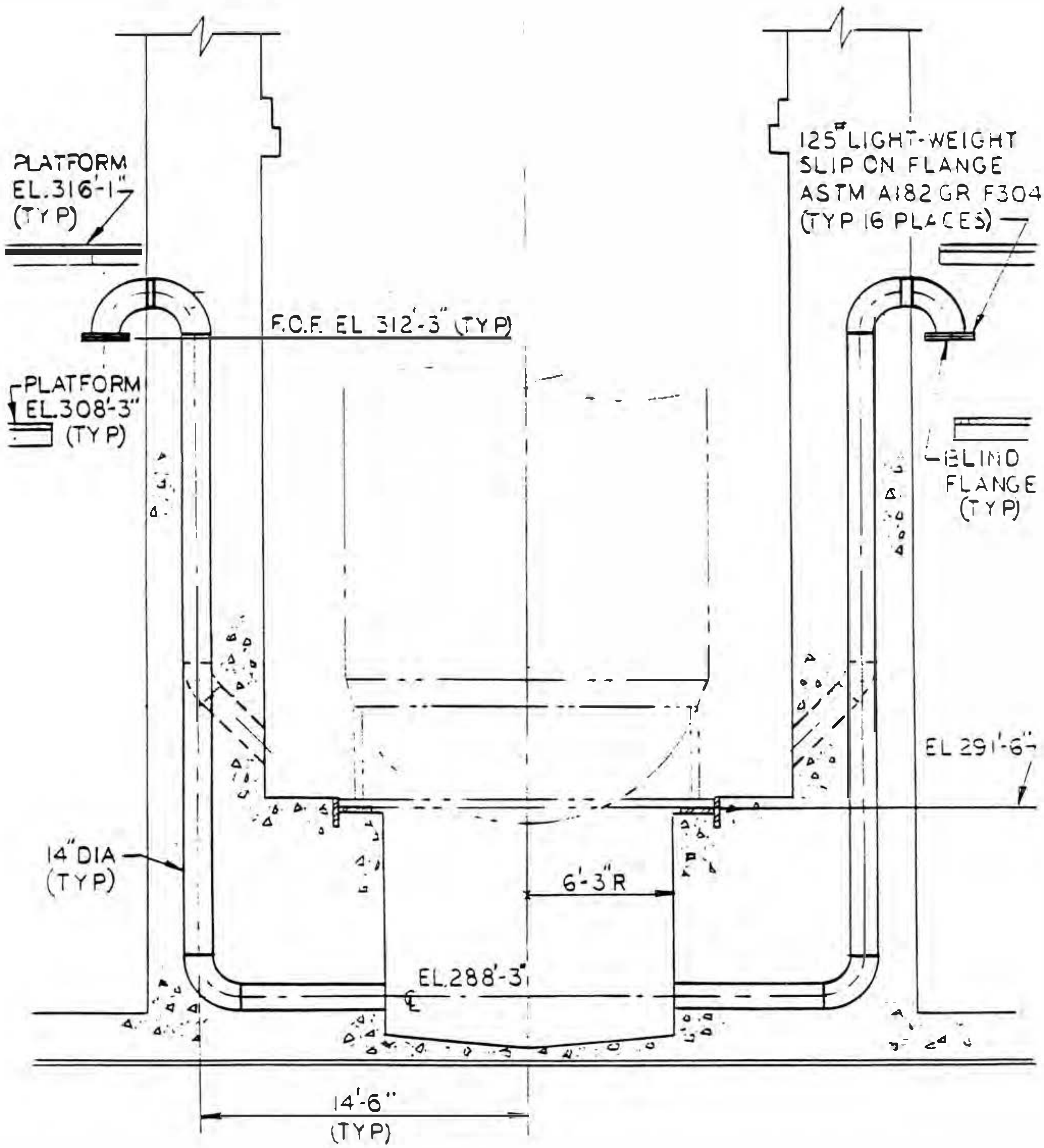
8.1 REACTOR VESSEL CAVITY AND LOWER RV HEAD

Since the canal seal plate was installed in the spring of 1984, the reactor vessel cavity has been assumed to be inaccessible. However, TAAG has identified another method of accessing this area.

The original concept for cooling the RV cavity consisted of sixteen 14-inch diameter ducts embedded in the primary shield wall. When the heat load was redefined during the construction effort, these ducts were abandoned in favor of the reactor vessel cavity supply fans (AH-E-52A&B). Since the ducts were embedded, they were not removed. Instead, the ducts were sealed outside of the primary shield. This accessible end is at El. 312', sealed with a 125# flange, and is accessible from the top of the D-rings.

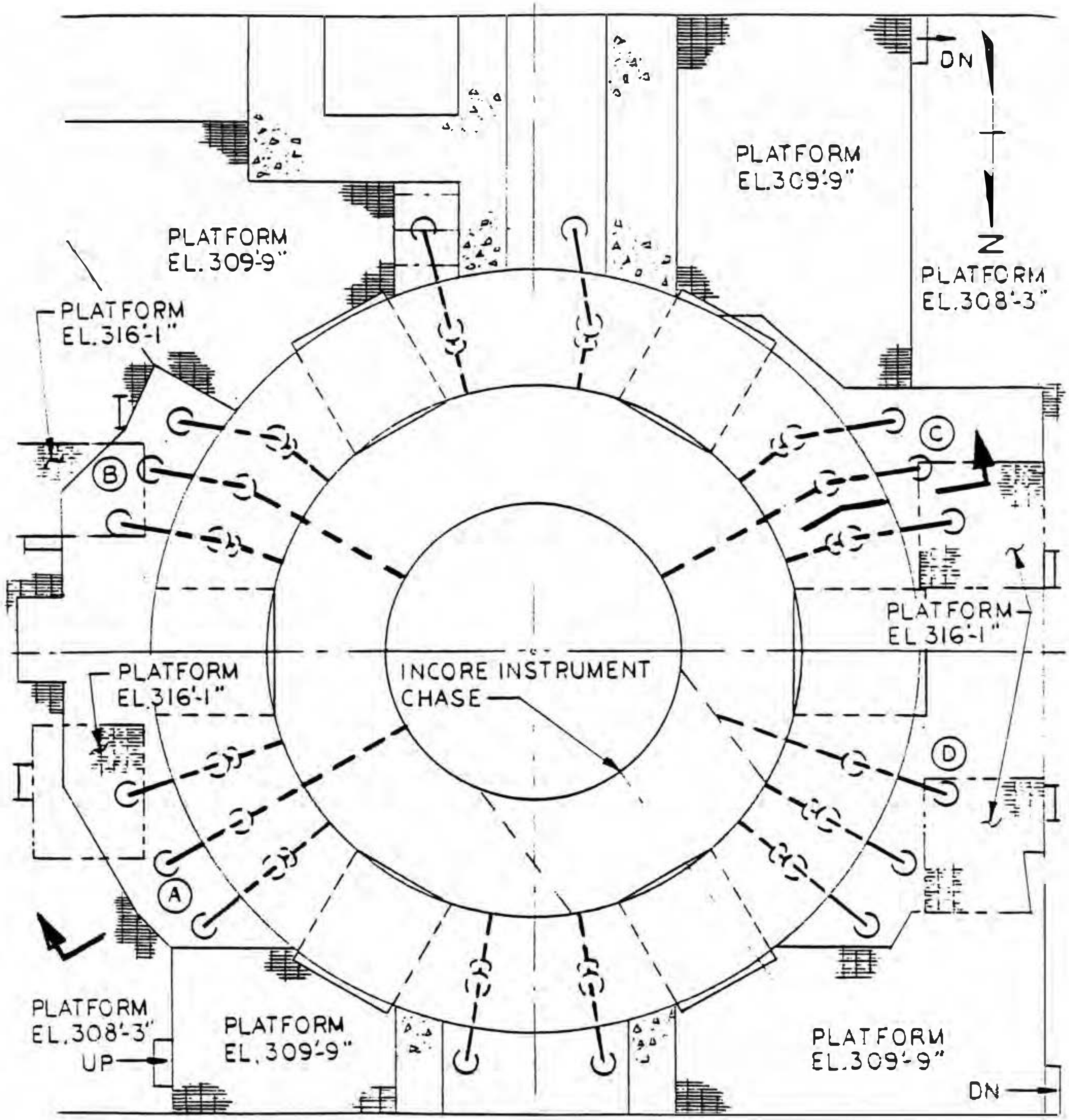
Twelve of these ducts terminate above the support skirt of the reactor vessel. Four of these ducts terminate at elevation 283'3", beneath the reactor vessel (see Figure 8-1). Three of these ducts terminate 6'3" from the centerline of the reactor vessel (labelled A, B, and C on Figure 8-2). The fourth (labelled D on Figure 8-2) is cut flush with the side of the in-core instrument chase.

These ducts could be used to introduce radiation instruments into the area below the lower reactor vessel head. The inside diameter of 14" schedule 30S pipe is 13.25 inches. This should be large enough to accommodate a wide range of instruments, including closed-circuit television cameras, without difficulty.



SECTION-R.V. CAVITY COOLING DUCTS

FIGURE 8-1



PLAN-PLATFORM DETAILS NEAR DUCTS

FIGURE 3-2

These ducts could provide a mechanism for refining the lower bound estimate of fuel debris on the bottom of the RV. This estimate was developed by HEDL using the solid-state track recorders (SSTRs). HEDL used the neutron flux from the lowest SSTR in the vessel cavity as the neutron flux beneath the reactor vessel. This was admittedly conservative in that it resulted in less predicted fuel. If a neutron detector were actually introduced into the area beneath the reactor vessel, that data combined with the SSTR analysis would raise the estimate of the lower bound for the fuel debris on the lower head.

Another possible examination using these ducts is to view the condition of the in-core guide tubes. Also, watching for leakage using CCTV cameras could provide a more direct indication than is currently possible in the unlikely event of small leaks from the bottom of the RV.

However, the access to the platform on elevation 308'3" is complicated by the fact that many of the ladders planned were not installed to expedite the completion of the plant's construction. The platform at El. 308'3" is accessible only from the basement with the currently installed ladders. Temporary ladders could eliminate this concern, but, due to the dose rates in the D-rings, would impose a worker exposure penalty while rigging the ladders in position. Also, dose rates at the accessible ends of the ducts are approximately 1R/hr. Hence, any data acquired via these ducts would be at a relatively large worker exposure. TAAG does not see any data acquisition, at present, worth the worker exposure involved with working in 1R/hr dose rates. However, if the need arises, the avenue exists to survey and to view the reactor vessel cavity beneath the lower head and the in-core guide tubes.

8.2 LETDOWN COOLERS

The letdown coolers are believed to contain fuel debris. Their position in the RC makeup and purification system upstream of the pre-filters MU-F-5A/5B, which had fuel debris on them, virtually assures that the letdown coolers contain fuel debris. However, these heat exchangers are located in a cubicle adjacent to the RB sump cubicle on El. 282'6". The dose rates in this area are too high to permit access with the necessary survey instrumentation. TAAG has identified a method of surveying this cubicle remotely from El. 305'.

Figure 8-3 shows the layout of the HVAC duct work in the RB basement. It can be seen from this drawing that the duct work is relatively level and that it enters into the letdown coolers cubicle. A relatively simple robot, similar to the type used to inspect gas lines, could be used to take a gamma spectrometer to the letdown coolers cubicle. The access to this duct is through the duct work attached to the east end of the RB cooling fans air plenum. One balance damper in the 20" duct (see Figure 8-4) must be either cut or negotiated by the robot, but no other obstacles are present.

This pathway is complicated but it may be the only ALARA way to obtain data about the contents of the letdown coolers in the foreseeable future.

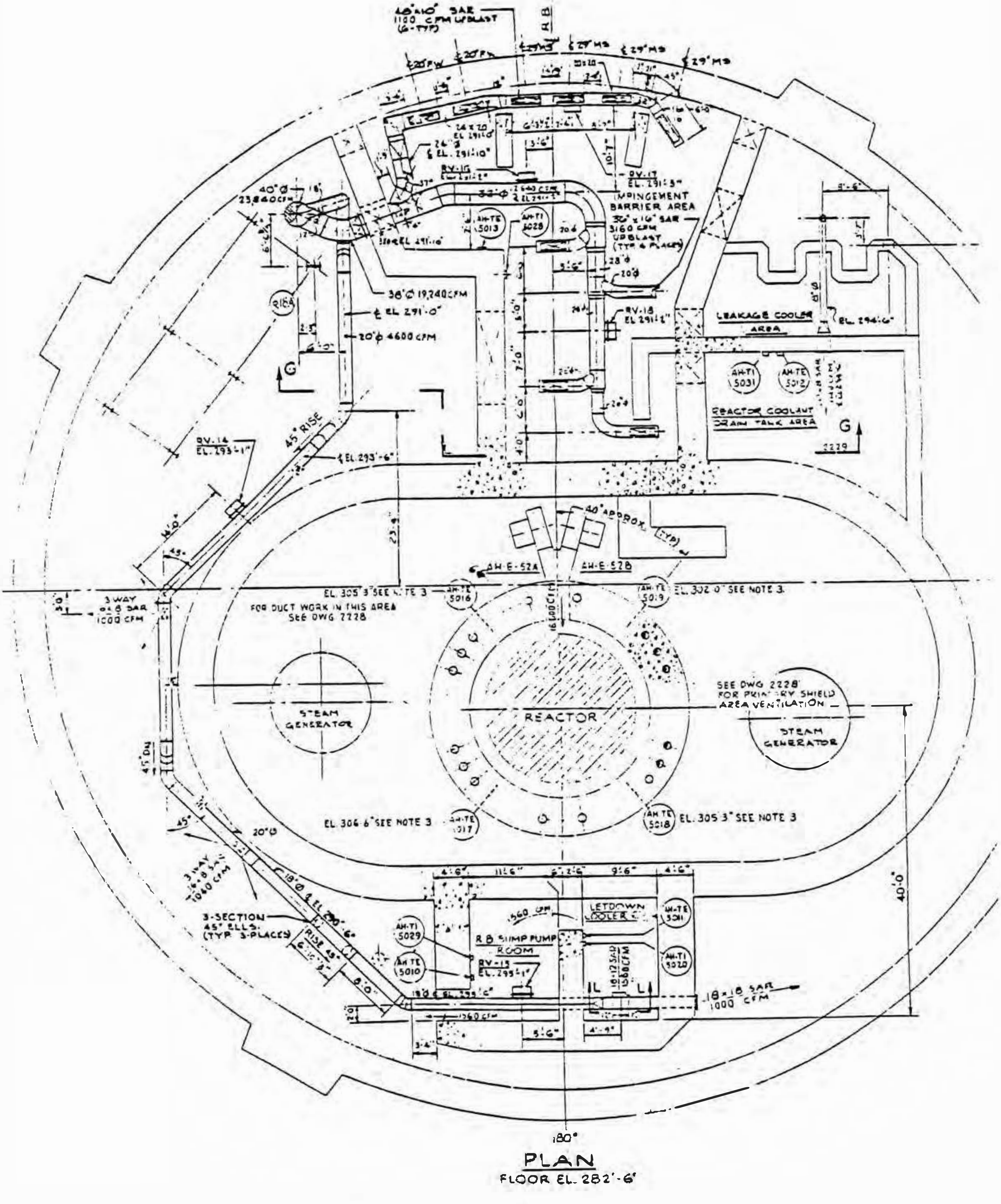


FIGURE 8-3
HVAC DUCT WORK IN REACTOR BUILDING BASEMENT

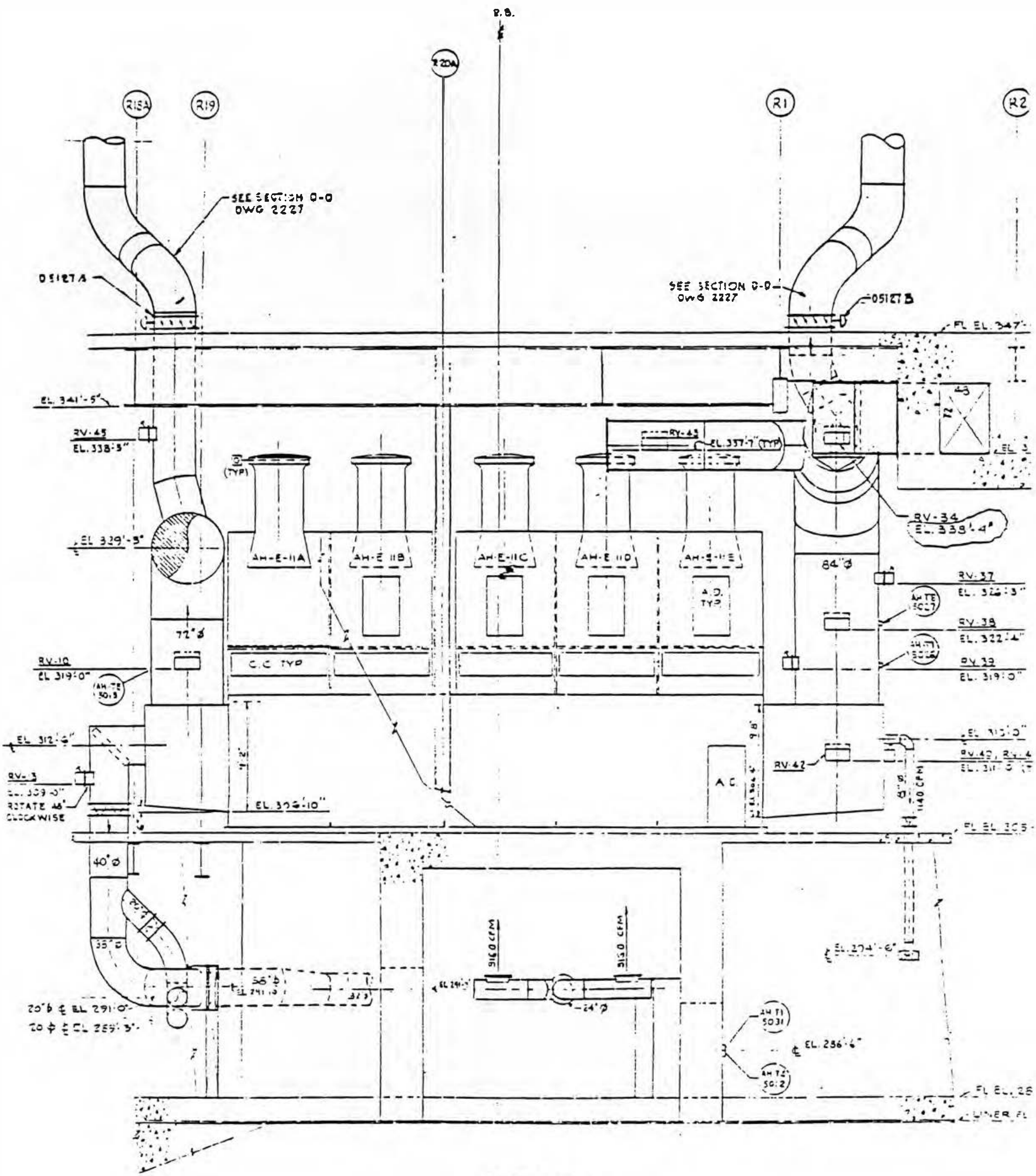


FIGURE 8-4
HVAC DUCT WORK ACCESS ROUTES

SECTION 9.0

DEFUELING WITH LEAKS

GPUN Site Engineering has made an assessment of the potential of leaks developing in the reactor vessel. This assessment indicates that the only credible reactor vessel leak (i.e., leakage of water below elevation 314') is from the 52 in-core instrument tubes that penetrate the bottom head of the reactor vessel. This assessment indicates a maximum leakage rate of 6 gpm per instrument tube and potentially up to about 300 gpm if all 52 tubes should fail. TAAG considers this GPUN assessment of potential reactor vessel leakage to be a "worst case" leakage scenario. Accordingly, this leakage scenario was used by TAAG in considering the question/issue of: "Evaluation as to whether defueling operations should continue in the event of leaks from the reactor vessel."

The GPUN reactor vessel leakage assessment also describes a system to maintain the reactor vessel filled in the event that such a leak develops. The system initially uses water from the BWST to keep the vessel filled by means of gravity and/or pumped flow of BWST water. For the long term, the water level in the reactor vessel is maintained during such a leak by collecting the leakage water in the reactor building sump and recirculating it back into the reactor vessel.

Accordingly, TAAG recommends that defueling operations can continue if such a recirculating system is available and it maintains an adequate shielding water depth in the reactor vessel. In this regard, it should be noted that all PWR commercial power plants have, in essence, such a controlled leakage system during defueling operation (i.e., reactor canal and RV water clean-up system). These systems take water out of the reactor canal and reactor vessel areas and then recirculate it back to reactor refueling areas. Functionally, fuel handling in the event of a leak is not different from the performance of the recirculation system in the event of RV leakage at any time.

SECTION 10.0

EQUIPMENT HATCH REMOVAL

TAAG was asked to evaluate the technical pros and cons and the safety aspects of removing the equipment hatch from the TMI-2 reactor building. This evaluation was to include the technical reasons for removing the equipment hatch, the technical problems associated with removing the equipment hatch, and the potential benefits that could be realized by removing the equipment hatch. The results of the technical review are presented below. As part of the safety review, a draft SER was prepared. It has been transmitted to GPUN under separate cover.

The equipment hatch is located in the southwest quadrant of the reactor building. It is a 23' diameter penetration in the reactor building wall and is provided to permit the movement of large objects into and out of the reactor building during an outage. A removable personnel air lock (air lock #1) is incorporated into the equipment hatch. Both the hatch and the air lock are double gasketted and bolted to steel flanges in the building. The seal is designed to withstand the effects of design basis accidents. Figure 10-1 shows a sectional view of the equipment hatch, the personnel air lock, and the missile enclosure.

The procedure for removing the equipment hatch requires that the personnel airlock be removed first. This airlock assembly can be withdrawn intact utilizing the monorail installed in the missile shield enclosure. The 9' outside diameter, 12'6" long airlock weighs 15 tons and is provided with lifting lugs to facilitate its removal. Once the personnel air lock is removed, the equipment hatch can be opened. The equipment hatch is 24'8"

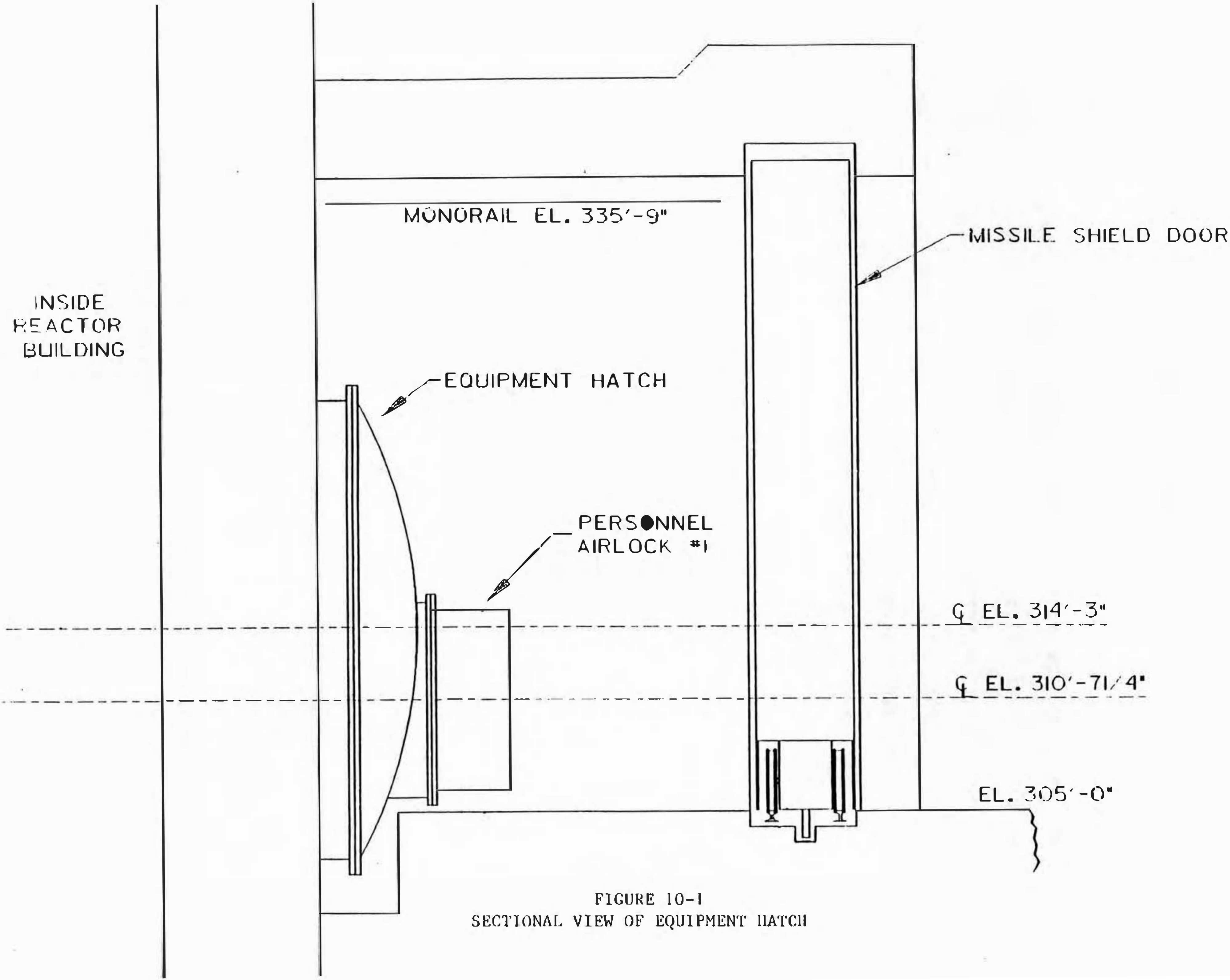


FIGURE 10-1
SECTIONAL VIEW OF EQUIPMENT HATCH

outside diameter and weighs 20.5 tons. It cannot be removed using the installed monorail because there is not adequate head room to permit the hatch to be lifted over the recess in the control building roof slab (see Figure 10-2). The equipment hatch has to be removed using cherry pickers or some other sort of temporary lifting equipment.

The reason for removing the equipment hatch is to permit the movement of large components and equipment into or out of the reactor building. Numerous large components, structures, or systems will soon need to be taken into the reactor building to support the defueling effort. Table 10-1 lists some of the more important of these. The present strategy is to carry these items into the reactor building in pieces via the personnel air locks, and to assemble the pieces inside containment. If the equipment hatch could be opened, many of these items could be assembled outside of the reactor building, tested, and brought into the building ready for installation. This strategy has many advantages:

1. Major construction activities can take place in a non-radioactive area using conventional construction techniques. This will enhance worker productivity for the construction efforts and will permit the use of non-RWP personnel, It will also eliminate the radiation exposures for the construction efforts. Since the construction can take place in several areas, major components can be built in parallel without competing for the available space inside the reactor building. The effect of constructing these components outside of the reactor building will be to accelerate the construction schedule

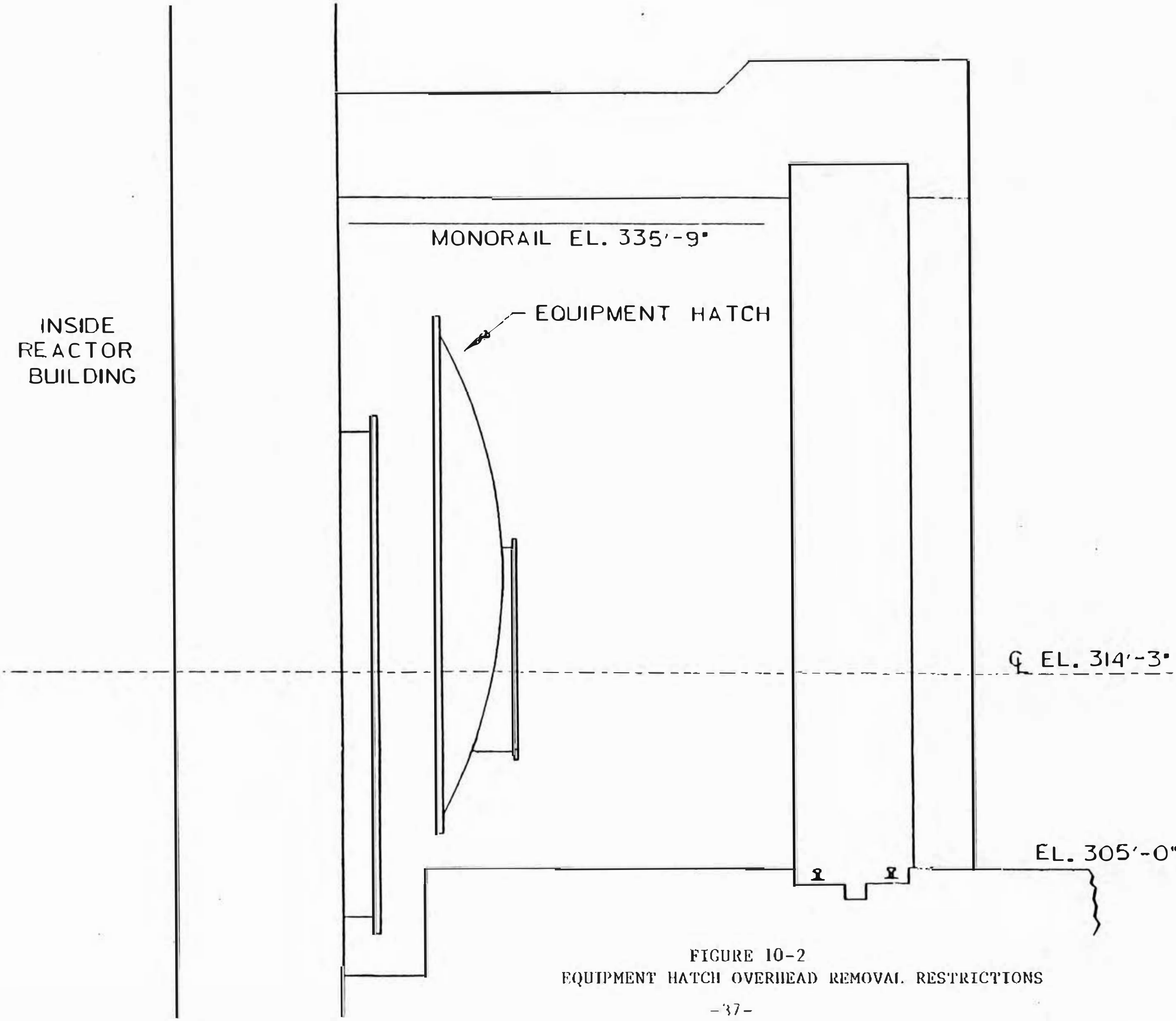


FIGURE 10-2
EQUIPMENT HATCH OVERHEAD REMOVAL RESTRICTIONS

TABLE 10-1

LARGE COMPONENTS REQUIRED FOR DEFUELING

Rotating Work Platform and Support Structure

Defueling Support Structure

Shielded Work Platform

Defueling Water Cleanup System

Dam for Deep End of Fuel Transfer Canal

Fuel Storage Racks for Canal

Defueling Canister/Tool Racks in Vessel

Gantry Crane

Defueling Canister Transfer Casks

Various Manual and Automated Defueling Tools because of improved worker efficiency and the possibility of parallel construction efforts, and to reduce the personnel exposures during the construction effort.

2. Components do not need to be designed to be assembled inside the reactor building. Opening the equipment hatch will reduce the engineering and design efforts required to design the items listed in Table 10-1 so that they can be carried into the reactor building in pieces. More conventional designs can be developed without concern for manually handling the individual pieces. This will permit better structural designs and lower costs for engineering and, possibly, for the final assembled item. This will have beneficial schedule and budgetary impacts.

3. Fewer entries will be required in the reactor building. Opening the equipment hatch will permit a single, although more complicated, entry to replace numerous entries required to carry in the pieces of the items listed in Table 10-1. Instead of carrying in dozens of pieces, the final assembled items can be efficiently brought into the building using heavy moving equipment. In addition, fewer tools, testing equipment, and equipment required for assembly will need to be staged into the building, thus eliminating the entries required to bring these items into and out of the building. This will result in less worker exposure, and less time and money spent on entry-related efforts. These have positive schedule and ALARA impacts.

4. The equipment brought into the reactor building can be pre-tested and pre-rigged. This will avoid start-up and operational testing in a

RWP area. This will permit better working conditions for calibration and through the repairs/modifications often required by start-up testing. Pre-rigged skids or assemblies will permit an approved lifting device to be verified outside of the reactor building. When the assembly is brought in, it can be immediately lifted off the transporter by the polar crane and, in most cases, moved to its installation staging area directly. This will improve the utilization efficiency of the polar crane and will reduce the number of lifts required. Together, the pre-tested/pre-rigged capabilities of using the equipment hatch will reduce radiation exposures to the workers and shorten the time required to get each system or component operational.

5. Large material handling equipment can be brought into the reactor building to assist in lifting and moving heavy loads. The size and shape restrictions imposed by the personnel air locks have prevented the use of large material handling equipment inside the reactor building. When the equipment hatch is open, large dollies, forklifts, or trucks can be brought into the building. These items can be left in the building once the equipment hatch is closed to assist material handling and to reduce the amount of manual labor and polar crane utilization. This could make the occasional use of the personnel air lock to bring in large or heavy objects easier on a limited basis. This will reduce the need to remove and replace the equipment hatch for every oversized item required for defueling. This will reduce worker radiation exposures and the level of effort required to carry heavy or bulky items into the reactor building.

In short, opening the equipment hatch seems to be a method of accelerating the defueling schedule, optimizing equipment design and testing, and reducing costs without either increasing worker radiation exposures or decreasing worker safety. Moreover, it appears to be an ALARA approach to defueling. To support this, TAAG has undertaken an ALARA estimate for opening the equipment hatch.

Table 10-2 shows the results of the estimate. All ALARA estimates are based on assumptions. This particular estimate is further hampered by the imprecise knowledge of the real scope and extent of work to be performed. For this reason, conservative assumptions were made to reduce the advantage of opening the equipment hatch. The amount of effort required to bring in pieces, to assemble components, and to startup and test systems in the reactor building have been minimized. Even so, the results indicate that the radiation exposure drops by a factor of six if the equipment hatch is used. While by no means definitive, the results of this scoping study ought to represent the minimum expected ratio between worker exposures with the equipment hatch open and with the equipment hatch closed. Actual savings are expected to be significantly larger.

There are three strategies for the removal of the equipment hatch:

1. Remove the equipment hatch in order to perform a specific activity and replace it when that activity is completed.

2. Remove the equipment hatch and replace it with a light-weight replacement hatch that is easier to open and to close.

TABLE 10-2

ALARA ESTIMATE FOR EQUIPMENT HANDLING IN THE REACTOR BUILDING

	No. of Entries		Average Dose Rate (mr/hr) in R.B. El. 305'-0"		Average Dose Rate (mr/hr) in R.B. for Construction and/or Testing		Man Hours per Entry		Man Hours for Construction		Man Hours for Testing		Average Entry Radiation Man-Rem		Average Area Radiation Man-Rem due to Construction and Testing		Total Man-Rem	
	1	2	3	4			5	6					Note 1	Note 2				
	Case 1	Case 2	Case 1	Case 2	Case 1	Case 2	Case 1	Case 2	Case 1	Case 2	Case 1	Case 2	Case 1	Case 2	Case 1	Case 2	Case 1	Case 2
To Support Plenum Removal																		
• Work Platform	50	1	175	220	50*	N.A.	1	2	50	N.A.	N.A.	N.A.	8.75	0.44	2.75	N.A.	11.50	0.44
To Support DWCS																		
• Piping Manifolds	12	1	175	220	50*	N.A.	2	2	12	N.A.	24	8	4.20	0.44	1.97	0.46	6.17	0.90
To Support Dry Defueling																		
• Dam for Deep End of Canal	1	1	175	220	50*	N.A.	4	2	1	N.A.	N.A.	N.A.	0.70	0.44	0.08	N.A.	0.78	0.44
• Support Structure	12	1	175	220	50*	N.A.	1	2	12	N.A.	N.A.	N.A.	2.10	0.44	0.65	N.A.	2.75	0.44
• Fuel Racks in Canal	10	1	175	220	50*	N.A.	4	4	32	N.A.	N.A.	N.A.	7.00	0.88	1.77	N.A.	8.77	0.88
• Canister/Tool Racks Vessel	6	1	175	220	50*	N.A.	1	2	6	N.A.	N.A.	N.A.	1.05	0.44	0.33	N.A.	1.38	0.44
• Mods for Main Fuel Handling Bridge	6	1	175	220	50*	N.A.	1	2	6	N.A.	4	4	1.05	0.44	0.56	0.23	1.61	0.67
• Rotating Work Platform	50	1	175	220	50*	N.A.	1	2	80	N.A.	40	N.A.	8.99	0.44	6.56	N.A.	15.55	0.44
To Support Complete Program																		
• Gantry Crane	5	1	175	220	50*	N.A.	2	2	5	N.A.	24	N.A.	1.75	0.44	1.59	N.A.	3.34	0.44
• Transfer Bell	5	1	175	220	50*	N.A.	2	2	5	N.A.	6	N.A.	1.75	0.44	0.61	N.A.	2.36	0.44
• RPV Gasket	1	1	175	220	50*	N.A.	1	1	N.A.	N.A.	N.A.	N.A.	0.18	0.22	N.A.	N.A.	0.18	0.22
• 50 Long Handled Tools	50	1	175	220	50*	N.A.	0.5	2	N.A.	N.A.	N.A.	N.A.	4.38	0.44	N.A.	N.A.	4.38	0.44
Totals							21.5	27	209	N.A.	98	12	41.90	5.94	16.87	0.69	58.77	6.17

Case 1 - ALARA for equipment handling in the reactor building using personnel air lock entry.
Case 2 - ALARA for equipment handling in the reactor building using equipment hatch entry.

* Exposure to workers during construction and testing is 50 mR/hr.
The dose for each entry and exit per worker is 14 mR in addition to stay time.

Note 1 - Column 1 x Column 3 x Column 5
Note 2 - Column 2 x Column 4 x Column 6

3. Remove the equipment hatch and construct an inexpensive confinement building to act as a low performance air lock for the rest of the recovery operation.

TAAG has reviewed each of these strategies and has concluded that the first is the easiest to justify technically. It is, however, not without drawbacks.

Opening the equipment hatch necessarily violates the integrity of the containment system. Due to the size of the hatch and complexity of removing and replacing it, the loss of containment will be in terms of shifts rather than minutes as is the case when both doors of an air lock are opened. The reactor building purge system can assure that the net flow air is into the reactor building, but it may not prevent the escape of some small amount of contamination due to the effects of winds. If a quick closing door (strategy 2) or a protective air envelope (strategy 3) were employed, they could reduce or prevent the escape of contamination from the building. Also, because of the absence of a dedicated equipment hatch handling system, there is the possibility of damaging the equipment hatch during its handling to the extent that a technical specification seal could no longer be achieved. The draft SER for temporarily opening the hatch, strategy 1 above, concludes that these operations can be conducted without undue risk to the health and safety of the public.

TAAG believes that strategy 1, i.e., remove the equipment hatch to perform a specific job or set of jobs, then replace the equipment hatch, is superior overall. The reasons for this are:

1. Removing and replacing the equipment hatch is a standard practice at all nuclear plants. It involves nothing new in the way of procedures or equipment.

2. The reduction in containment integrity is restricted to the period of time when the equipment hatch is off. Fire watches and other procedural changes need not be evoked after the equipment hatch is replaced. Hence, worker radiation exposures and schedule impacts of these extraordinary precautions need not affect defueling or other in-containment activities.

3. Prudent handling and storage procedures can be employed to reduce the risk of damaging the equipment hatch.

4. No engineering, design, procurement, or construction efforts need to be undertaken to substitute another structure for the equipment hatch.

In summary, TAAG sees compelling reasons to open the equipment hatch. They are:

1. Reduction of radiation exposures

2. Increased efficiency of workers and of other resources

3. Acceleration of defueling schedules

4. Improved quality of equipment used for defueling

5. Reduced chance of injury and of accidents in the reactor building.

TAAG further recommends that the equipment hatch be removed to perform a job or set of jobs and then be replaced. Large items should be staged outside the reactor building as much as possible to reduce the number of times and the duration of time that the equipment hatch is removed. Special measures should be taken to reduce the environmental risks associated with opening the equipment hatch, and these measures should only be evoked while the equipment hatch is open.

SECTION 11.0

FISSION PRODUCT BEHAVIOR IN FUEL DURING DEFUELING*

TAAG developed a method for estimating fission products released by mechanical disruption of fuel during TMI-2 defueling operations. The approach is: 1) estimate the quantities of fission products of interest that were released during the course of the accident; 2) estimate the manner in which the remaining quantities are currently distributed within the fuel; 3) evaluate releases of the remainder as fuel is crushed (i.e., as new surfaces are created) in the course of defueling; and 4) evaluate releases due to subsequent leaching of fission products from the new surfaces.

This treatment does not consider the production, composition, and behavior of fuel or other solids fines. Therefore. only five radionuclides are of interest: Kr-85, Cs-134, Cs-137, Sb-125, and Sr-90.

A method for estimating fission product release during defueling has been reported in TPO/TMI-133. After estimating fission products currently distributed within the fuel, releases were evaluated 1) as the remainder of the fuel is "crushed" and new surfaces are created during defueling, and 2) by subsequent leaching. The results presented in TPO/TMI-133 may be used for planning defueling water cleanup system operations.

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* This section summarizes TPO/TMI-133.

11.1 FISSION PRODUCTS RELEASED DURING THE TMI-2 ACCIDENT

The inventory of fission products released at the time of the accident was calculated via ORIGEN 2. Also, air and water samples have been taken intermittently, since the accident, and these data were used to estimate the quantities of krypton, cesium, and strontium which have escaped from the core since March 28, 1979. Antimony release estimates were derived from laboratory-scale, hot-cell studies of the relative rates of escape of Sb-125 and Cs-137 from irradiated fuel specimens.

11.2 CURRENT DISTRIBUTION OF FISSION PRODUCTS WITHIN THE TMI-2 FUEL

The values for the initial fission product inventories, the estimated amounts released, and the known half-lives were then used to calculate the current inventories of the fission products of interest.

Fission product release from UO_2 under light water reactor conditions (i.e., in a hydrogen-rich, steam atmosphere) is inconsequential at temperatures below about 1000^oC (except for the release of the so-called "gap-inventory"). Thus, since some analyses of the temperature history of the core during the accident suggest that part of the core remained covered with water, the current inventories of the fission products are probably not distributed uniformly throughout the fuel in the reactor vessel. Although the final results are not substantially affected by the precise manner in which the fission products are distributed throughout the fuel, the following assumptions were, nonetheless, made:

1. Two-thirds of the original core has been reduced to a debris bed, whereas the remainder may be regarded as comprising "intact" fuel pellets.

2. All of the fission products that escaped the core during the accident were uniformly released from the fuel which currently makes up the rubble bed, whereas none of the fission product inventories of the intact fuel pellets was released.

11.3 RELEASE RATES AS FUEL IS CRUSHED

Calculations were performed to represent the releases anticipated after <u>all</u> of the two fractions of fuel have been pulverized. These estimates will be refined once a detailed defueling plan has been developed.

11.4 RELEASE RATES AS A RESULT OF LEACHING

Once the fuel has been crushed, additional leaching of the fission products of interest will occur as the fuel particles contact the water. The procedures and results described by Mitchell, Goode, and Vaughen (ORNL/TM-7546, May 1981) were used to calculate these leach rates.

SECTION 12.0

BORONOMETER

TAAG was asked to provide information on a contingency boronometer including vendor, cost, operating experience, and why it is better than the existing instrument.

It is anticipated that this device will be used to monitor in-core boron concentration during defueling. GPUN requested this information because it was having difficulties with its existing boronometer. Prior to looking into this matter, TAAG had stated that the existing instrument should be satisfactory, the concept of a boronometer is simple and the existing unit had a good reputation. Subsequent investigations by TAAG confirmed that no available boronometer was better than the one GPUN already owned.

Subsequent GPUN efforts corrected the problems it was having with the existing boronometer. These problems were associated with moisture build-up as a result of long-term storage and calibration errors.

As a result of its efforts relating to the boronometer, TAAG suggested that GPUN evaluate the feasibility of using conductivity meters to monitor the boron concentration. Preliminary studies by GPUN have demonstrated the ability of these devices to provide the measurements with the required accuracy.

DISPOSITION OF TAAG RECOMMENDATIONS FROM THE EIGHTH TAAG REPORT

RECOMMENDATION I.D.	RECOMMENDATION	DISPOSITION
1.2.1/1-page 2	Terminate work on the decon skid and use subcontractor.	GPUN has terminated such work.
1.2.1/2-page 2	CADD decontamination work should be terminated.	The CADD is proving useful for this work. It will continue to be used.
1.2.1/3-page 2	AFHB work should be re-prioritized.	AFHB work is being re-prioritized consistent with the Strategy Plan.
1.2.1/4-page 2	Procedural documents should be prepared using original terminology.	Cubicle designations are understood by on-island personnel.
1.2.4 -page 3	Utilize a defueling concept that incorporates dry transfer from the reactor vessel to the deep end of the transfer canal.	GPUN plans are now similar to TAAG's recommendation.
1.2.5 -page 3	The PEIS supplement should include comparisons of exposure with other cases of radiation exposure.	NRC only addressed this subject in the PEIS supplement by health risk estimation.

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