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June 18, 1984

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

Dear Dr. Snyder:

Three Mile Island Nuclear Station, Unit 2 (TMI-2)
Operating License No. DPR-73
Docket No. 50-320
Safety Evaluation Report for Preparatory Activities for
Plenum Assembly Removal

Attached for your review and approval is the Safety Evaluation Report (SER) for activities preparing for plenum removal. This SER addresses activities such as visual plenum inspection, measurement of interference gaps, cleaning of the underside of the plenum and potential interference areas, and attempted movement of the Axial Powered Shaping Rods (APSR's). The purpose of these activities is to determine if potential interference may hinder removal of the plenum and to remove debris that may cause binding during plenum removal. It has been determined that the described activities can be accomplished without undue risk to the health and safety of the public. This activity has been determined to constitute a Class III Amendment, per 10 CFR 170.22. Thus, enclosed is a check for $4,000.00.

If you have any questions concerning this information, please call Mr. J. J. Byrne of my staff.

Sincerely,

B. K. Kanga

B. K. Kanga
Director, TMI-2

D009 w/check $4,000
00105473
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BKK/RBS/jep

Attachment

cc: Deputy Program Director - TMI Program Office, Mr. L. H. Barrett

SAFETY EVALUATION REPORT

FOR PREPARATORY ACTIVITIES FOR

PLENUM ASSEMBLY REMOVAL

FOR

GPU NUCLEAR

THREE MILE ISLAND - UNIT 2

Rev. 2
June 1984

SHEET	LATEST REV.	SHEET	LATEST REV.	SHEET	LATEST REV.	SHEET	LATEST REV.	SHEET	LATEST REV.	SHEET	LATEST REV.	SHEET	LATEST REV.
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REVISION STATUS SHEET

DOCUMENT TITLE: Safety Evaluation Report for Preparatory Activities for Plenum Removal for GPU Nuclear Three Mile Island - Unit 2

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1.0 Introduction

1.1 Background

The removal of the plenum assembly (PA) follows removal of the reactor vessel (RV) head and the installation of the internals indexing fixture (IIF). Removal of the PA is necessary to gain access to the core region for eventual core defueling. The PA was exposed to temperature extremes during the accident which could possibly have caused thermal distortion of the PA. In addition, fuel damage during the accident has resulted in the deposition of fuel fines and other radioactive material throughout the PA structure. Thermal distortion and debris deposition, either individually or in combination, may potentially interfere with PA removal. Potential interference points have been identified and are to be inspected prior to any attempt to lift the PA. Other preparatory activities for PA removal include the separation of unsupported end fittings, video inspection of the core void space, video inspection and movement of the axial power shaping rods (APSR's), and cleaning of the plenum.

1.2 Purpose

The purpose of these preparatory activities is to confirm the adequacy of plenum removal equipment and techniques. The purpose of this Safety Evaluation Report (SER) is to describe these preparatory activities and the tools that will be employed; to present what information is expected to be obtained; and to evaluate these activities to assure that this task can be accomplished without presenting undue risk to the health and safety of the public.

1.3 Scope

This SER covers only the preparatory activities associated with, and supportive of the PA removal. The sequencing of these activities is after the activities scoped by Reference 1 and prior to the initial lift of the PA.

1.4 Organization

Section 2.0 consists of the descriptions of the preparatory activities and the tooling and techniques employed. The information or results that are expected to be obtained is also addressed.

Section 3.0 consists of the radiological considerations associated with the planned activities including an assessment of expected external and internal occupational exposure and a presentation of the measures to be taken to reduce the occupational exposure to as low as is reasonably achievable (ALARA).

Section 4.0 addresses the safety concerns associated with the planned activities and evaluates postulated accidents. Also included is the evaluation with respect to 10CFR50.59.

Section 5.0 presents the conclusions of this SER.

2.0 Description of Activities

Preparatory activities for plenum removal would not commence until the following in-containment conditions are met.

- o The reactor vessel head is removed and parked on its storage stand at the 347'-6" elevation.
- o The IIF is installed and filled to approximately 5 feet with water having a boron concentration satisfying Reactor Coolant System technical specification and operating procedural requirements. Its cover (which may have up to 1 inch of lead for shielding) is installed to be used as a working platform.
- o The auxiliary fuel handling bridge is refurbished to be used as a working platform.
- o A crane is available for use.
- o IIF ventilation/filtration system (if required) is operational.
- o Interior IIF lighting system which is suspended from the IIF platform is operational.
- o IIF processing system is operational.
- o 480-, 208-, and 120-volt AC power available in the reactor building via portable power distribution centers.

In addition to the conditions given above, actions specified in Section 4.4 of Reference 1 and Section 3.3 of Reference 2 will remain in effect to monitor reactor coolant system (RCS) boron content and to preclude inadvertent boron dilution of the RCS.

The preparatory activities include:

- o video inspection of potential interference areas,
- o video inspection of the core void space and the APSR assemblies (APSRA's),
- o measurement of the LOCA restraint boss gaps,

- o measurement of the elevations of the APSRA's,
- o movement of debris by mechanical means via probing tools, brushes, and scrapers to improve inspections of, or to clean, potential interference areas,
- o separation of unsupported fuel assembly end fittings, and
- o attempting movement of the APSRA's.

Additional activities may be identified during inspection. These activities which are not bounded by this safety evaluation, will be evaluated prior to implementation. 2

Figures 2-1 and 2-2 depict the plenum assembly and the control rod guide assembly (CRGA)/fuel assembly interface, respectively. The plenum assembly is located directly above the reactor core and consists of a plenum cover assembly (which is a series of 2-inch thick ribs intersecting to form a square lattice with a perforated top plate and flange), an upper grid rib section (which is a rolled plate with machined holes that locate the lower end of the individual control rod assembly (CRA) guide tube assembly relative to the upper end of the corresponding fuel assembly), control rod guide assemblies (CRGA's) (which are 8-inch schedule 40 pipe welded to the plenum cover plate and welded to 3/4-inch thick mounting flanges that are bolted to the upper grid assembly), and a flanged plenum cylinder (which is 1 1/2 inches thick and 130-inch outer diameter with six 34-inch diameter holes and four 22-inch diameter holes for coolant flow) connected to the plenum cover plate flange and the upper grid.

2.1 Inspections

Seven interfaces have been identified as potential interference areas which require examination prior to an attempt to lift the PA. The following four interfaces will be inspected, as warranted and accessible, to evaluate the possibility of binding which could prevent lifting the PA. These four interfaces are shown on Figure 2-3 with the tag numbers corresponding to the descriptions given below.

1. Plenum Keyways - Equally spaced around the plenum support flange are four keyways which mate to their corresponding keys located on the RV flange. Circumferential deformation of the plenum support flange may cause binding at the key-to-keyway pinch points. Such binding could increase the loading required for a successful plenum lift.

2. LOCA Restraint Bosses - On each plenum hot leg are thirteen LOCA restraint bosses which align themselves with their counterparts located on the core support shield. Minor deformation of the plenum cylinder would cause a reduction of the gaps between some pairs of LOCA bosses which, in turn, could cause interferences at those locations.

3. Upper Grid Ring Forging/Core Support Shield Bottom Flange Interface - This interface is shown in detail on Figure 2-4. An interference may exist at this interface because of the potential for a large and, most likely, nonsymmetrical deformation of the upper grid rib section, and radial deformation of the upper grid ring forging.

4. Vent Valves - There are eight vent valves located around the core support shield cylinder whose innermost protruding parts may interfere with PA removal. The potential interference areas are shown in greater detail on Figure 2-5 and are described below.

   - Minor deformation of the plenum cylinder may cause an interference between the LOCA restraint bosses and the vent valve assemblies.

   - Radial deformation of the upper grid ring forging may cause an interference between it and the vent valve assemblies.

The following three interfaces will be inspected, as warranted and accessible, since they may not only interfere with the plenum lift but may also allow fuel or control materials to be lifted with the PA. These three interfaces are depicted on Figure 2-2 and are described below.

5. Upper End Fitting Ears - Each fuel assembly end fitting has eight ears (a pair on each side). The ears axially engage with the grid pads which are bolted and doweled to the underside of each grid rib. The amount of engagement is a function of temperature (thermal growth of the fuel assembly) and fuel burnup (elongation of the zirconium crystalline lattice due to irradiation). Deformation, debris accumulation, or silver soldering may prevent the separation of the fuel assembly end fitting from the plenum assembly.

6. Adjacent Fuel Assemblies - The clearance between the upper ends of adjacent fuel assemblies may have been reduced because of deformation or debris accumulation. Loss of this clearance could prevent a successful lift of the PA if the fuel assemblies cannot be separated from the PA.

7. CRA's and APSRA's - The individual rods in the CRA's and in the APSRA's are guided through their respective assemblies via split tubes and C-tubes. These thin-walled tubes, because of thermal distortion, may inhibit movement of the rods.

In addition, the seating surfaces on which the four lifting jacks will bear during the initial lift of the PA will also be examined. One end of the jack will bear on the underside of the plenum rib while the hydraulic piston foot end will bear against the exposed top surface of the core support shield top flange.

2.2 Measurements

The measurements performed during the preparatory activities include the gaps between the LOCA restraint bosses and the elevations of the APSRA's. A specially designed tool will be used to measure the gaps between the LOCA restraint bosses. These measurements will indicate the magnitude and direction of the thermal distortion in the plenum and core support shield cylinders. If required, all of the LOCA restraint boss gaps will be measured. A graduated tool will be used to measure the elevations of the APSRA's.

2.3 Manipulations

The manipulation of components within the plenum assembly will consist of dislodging unsupported fuel assembly end fittings and attempting movement of the APSRA's. All unsupported end fittings will be dislodged to drop onto the debris bed. Impact tools will be used to transmit the force required to free stuck end fittings.. An impact tool will be coupled to the APSRA's spiders to attempt upward (with the crane) or downward (with a sliding hammer) movement of the APSRA's. If capable the APSRA's will be driven into the core. Previous attempts to fully insert the APSRA's via their control rod drive mechanisms have been unsuccessful for six out of eight APSRA's.

2.4 Accessibility and Cleaning

Primary access to the reactor internals is through the control rod guide tubes (CRGT's) and around the periphery of the plenum cover between the ribs. The CRGT's provide direct access to the tops of control rods and fuel assemblies over which they are located when the leadscrews are removed. The CRGT brazement limits the size of tools to less than 2 3/8 inches in diameter. The flow holes located above the lower two spacers on the CRGT provide potential access to adjacent fuel assemblies. Removing unsupported end fittings and control rod spiders will provide direct access to the core cavity. The annulus between the support shield and the plenum cylinder provide a 2 3/4-inch opening for direct line-of-sight access to the vent valves, LOCA restraint bosses, hot legs, and lower flanges. Between the plenum cylinder and the cover plate direct vertical access is limited to 1 1/4 inches. However, because of the 10 3/4-inch vertical offset between the cover plate and the start of the cylinder, tools up to about 10 x 10 x 6 inches can be lowered into the interior of the plenum assembly.

Cleaning of the horizontal surfaces of the plenum assembly will be accomplished to the extent necessary for gaining accessibility. The surfaces to be so cleaned include the cover plate, ribs, support ring, the top flanges of the plenum and core support cylinders, the 69 control rod guide assemblies, and the topmost spacers.

2.5 Tooling

Remote tooling has been designed to accomplish the various preparatory activities. The tooling can be grouped into four categories based on their function: inspection, measurement, impact, and cleaning tooling. The tooling associated with each of these categories are described below. Supportive equipment for tool operation consists of tool counter balances, a load cell, temporary tool racks, a permanent tool storage rack, and a tool strong back for upending assembled tools.

2.5.1 Inspection Tooling

Video camera(s) will be used for visual inspections to assess selected areas of the reactor internals. They are manipulated with either a camera cable and draw wire or a camera handling tool assembly. The cable-hung camera hangs freely from its cable with a draw wire attached which enables positioning of the camera in areas inaccessible by a pole. The tool handle is provided with cut-outs for routing of the camera cable and draw wire and a pulley assembly for the draw wire. The camera handling tool assembly consists of the tool handle, a camera handling tool end, camera winch assembly, camera cable, and a camera. A pivoting camera clamp is welded to the bottom of the tool end and the camera winch assembly mounts on the tool handle and connects to a clamp mounted on the camera.

2.5.2 Measurement Tooling

Measurement tooling has been designed to measure the gaps between the LOCA restraint bosses and the elevations of the APSRA's. The LOCA Boss Gap Measurement Tool Assembly consists of a tool handle, a tool end, a Rees R-93 camera with a fixed right angle lens and a scale positioned in front of the Rees R-93 camera right angle lens. The elevations of the APSRA's are measured by using the APSRA Depth Gauge/Handling Tool Assembly (described below) in conjunction with the Gauge Sighting Device Assembly which is a leveling/sighting device welded to a rail clamp.

2.5.3 Impact Tooling

A variety of impact tools have been designed to attempt movement of the APSRA's and to ensure separation of all unsupported end fittings.

The APSRA Depth Gauge/Handling Tool Assembly consists of a tool handle, a tool end, and a slide hammer assembly. The tool end consists of an APSRA male coupling which inserts into the APSRA female coupling (spider). A 45° clockwise turn engages the tool with the APSRA spider to allow attempting of upward movement with a crane hoist. A 5,000 pound capacity load cell is attached between the tool assembly and the hoist hook to monitor the amount of force the crane hoist is applying to the tool. The slide hammer assembly is a slide hammer which weighs approximately three pounds and travels along a four inch slide which is mounted on the tool handle.

The Fuel Assembly End Fitting Probe Assembly consists of a tool handle, a tool end, and a slide hammer assembly. A nine inch tip is fitted and welded to the bottom of the tool end. The slide hammer assembly is identical to the one described for the APSRA Depth Gauge/Handling Tool Assembly. This is a low impact tool which may be used to separate end fittings from the plenum.

Three styles of End Fitting Separation Tooling have been designed to accommodate the different geometry and space constraints to access the fuel assembly end fittings and to supply more impact force than the Fuel Assembly End Fitting Probe Assembly. Each tool has an identical manual impact ram assembly which consists of a stainless steel shaft inside a section of stainless steel tubing. The Straight Tool is capable of being inserted between the plenum cover plate and plenum cylinder wall to impart an impact load on the top of the fuel assembly end fittings. The Offset Tool has the same insertion capability as the Straight Tool but the offset allows accessiblity to the ten core locations at the Z and W axes that are not accessible with the Straight Tool. The H-Tool is capable of being inserted down a CRGA, to dislodge the end fitting directly beneath it, and of being moved radially outward through the CRGA flow holes, to impact the four adjacent end fittings.

2.5.4 Cleaning Tooling

A variety of tools have been designed to perform the various cleaning activities required to access, inspect, and assess plenum areas.

The Debris Probe Tool Assembly consists of the tool handle and a debris probe tool end. The tool end is a section of stainless steel rod with the bottom end bored out to accommodate a stainless steel point which is welded in place.

An additional tool consists of a three section tool handle, lifting bale, and four interchangeable tool ends which are assembled using ball lock pins. The four tool ends include two stainless steel wire brushes of different sizes and a horizontal and a vertical scraper. Each tool has been designed for use on certain plenum surfaces for timely and efficient cleaning.

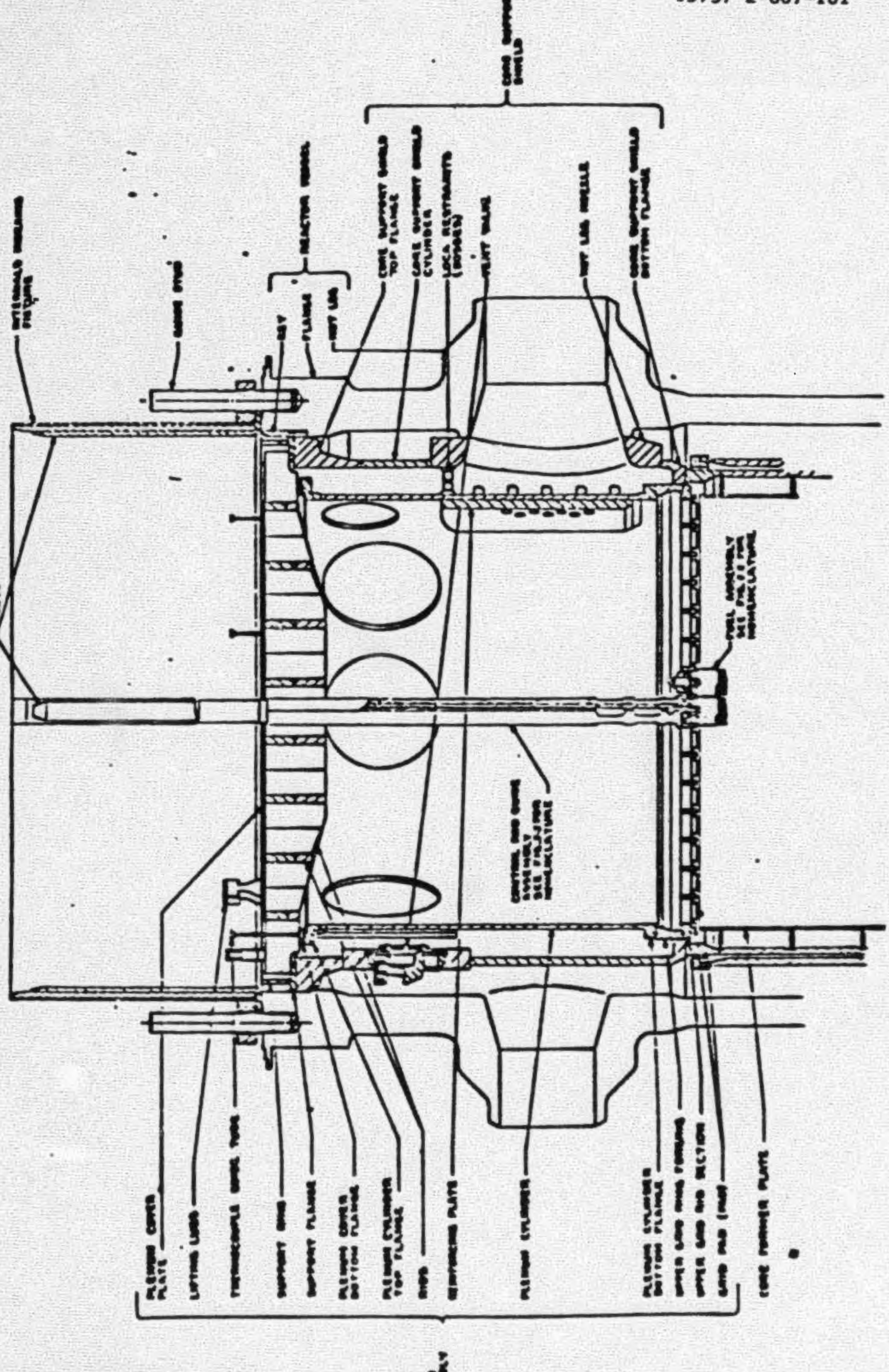


Figure 2-1
Upper Reactor Internals
Nomenclature

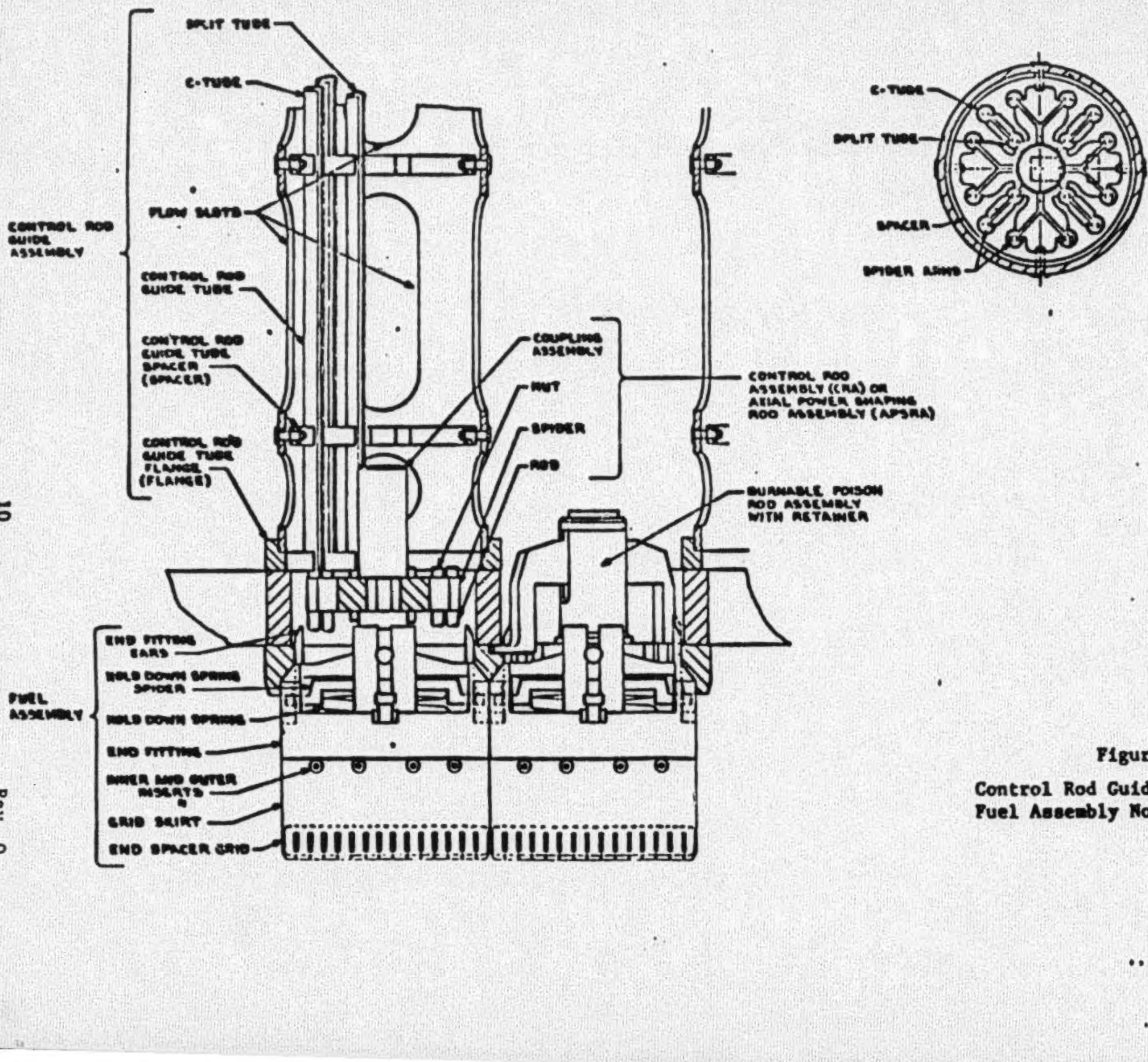
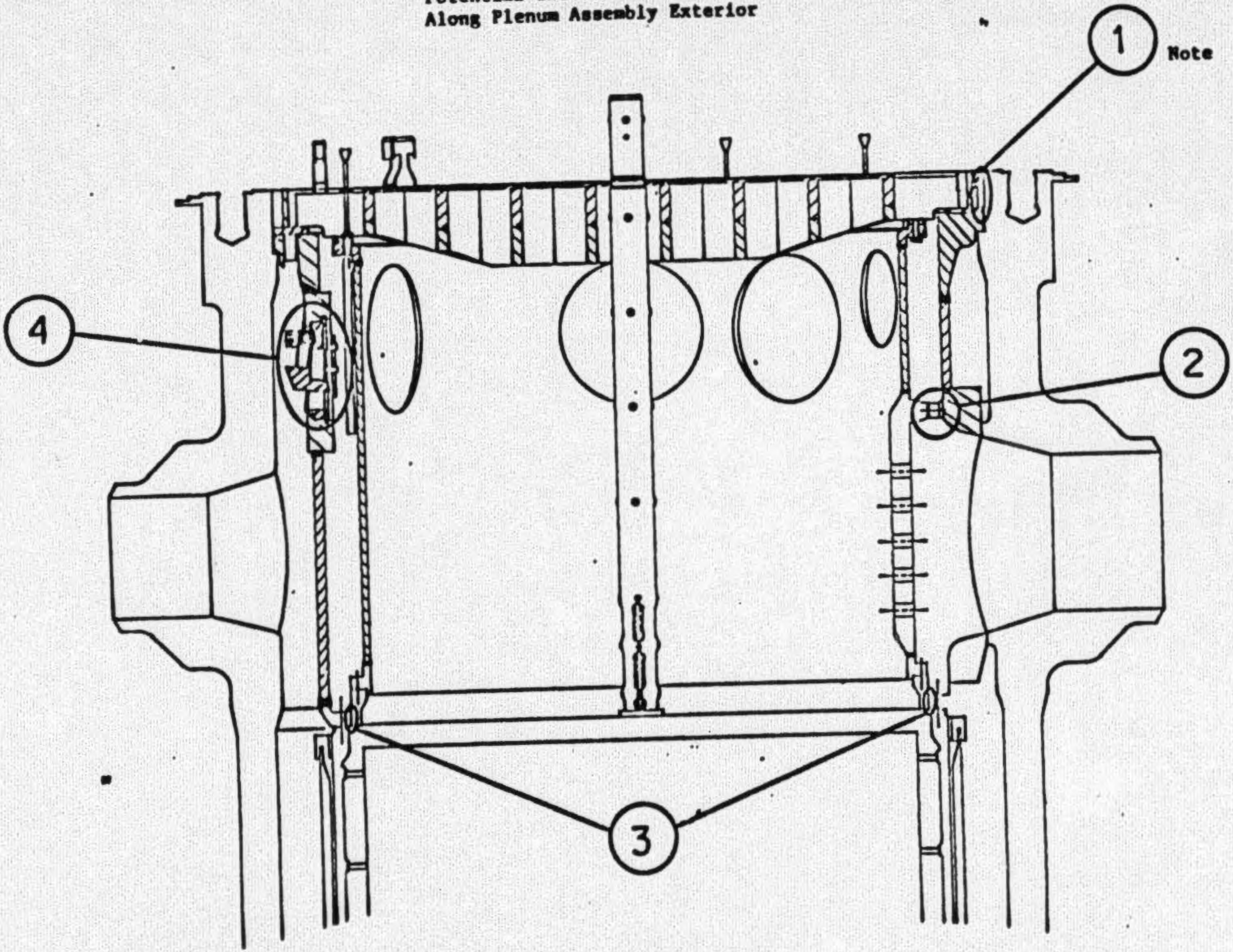


Figure 2-2

Control Rod Guide Assembly
Fuel Assembly Nomenclature

Figure 2-3

Potential Interference Areas
Along Plenum Assembly Exterior



Note: Numbers correspond to descriptions given in Section 2.1 [1]

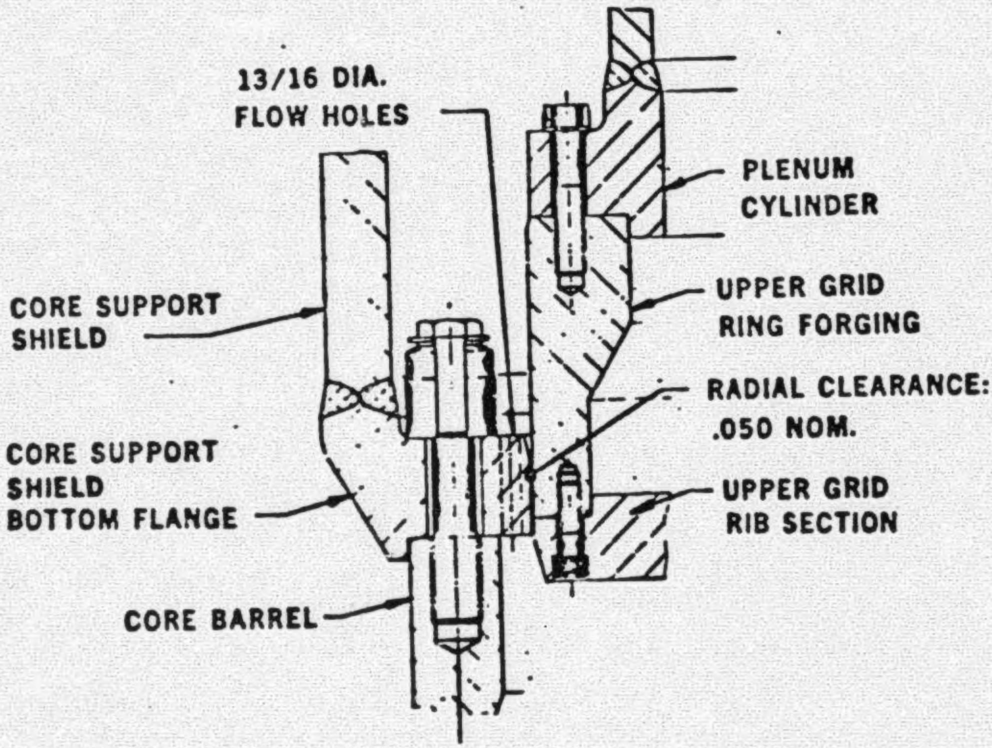


Figure 2-4

Upper Grid Ring Forging to Core Support Shield Bottom Flange Interface

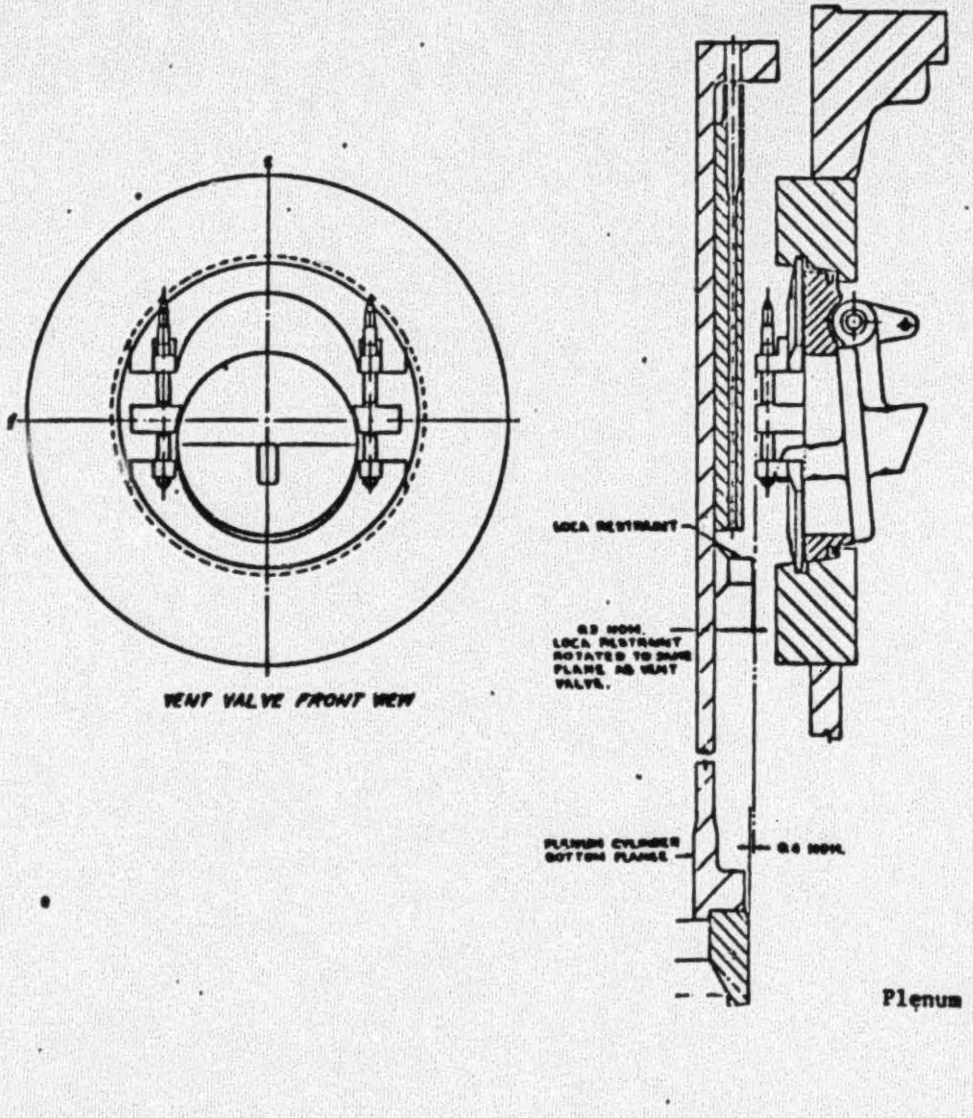


Figure 2-5
Plenum Assembly to Vent Valve Interface

3.0 Radiological Considerations

3.1 External Exposures

All individuals entering the reactor building will be monitored for external exposures in accordance with radiological control procedures to ensure personnel exposures are maintained as low as is reasonably achievable (ALARA) and within 10CFR20 dose equivalent limits. Administrative control points in accordance with the procedures will be used in order to assure specified dose limits are not exceeded. Extremity monitoring will be performed as needed in accordance with existing procedures. Radiological Controls Department personnel will continiously monitor dose rates in the reactor building during these preparatory activities for plenum assembly removal.

The collective personnel radiation exposure to workers during these inspection activities has been estimated. The estimate was developed based on projected person-hour requirements and reactor building exposure rates associated with these inspection activities. The person-rem estimate of 105 person-rem is taken from reference 3, which is based on 1035 in-containment person-hours.

Person-rem for radiological controls support is not included in the above estimate. From a review of historical data it is assumed that person-rem for the radiological controls group will be 20% of that accumulated by other groups in containment. Based on this, the estimate for radiological controls support is 21 person-rem, and the total for all groups is estimated at 126 person-rem.

Due to the uncertainty in the person-hour estimate and the radiological conditions which will exist during the inspection activities, it is estimated that the total exposure could vary by up to $\pm$ 30 percent. Considering these uncertainties, 90 to 165 person-rem has been selected to be used as the estimate for the performance of the preparatory activities for plenum assembly removal, including radiological controls support. Detailed exposure estimates will be developed on a task-by-task basis as a normal part of ALARA review of in-containment work and to ensure that each activity is performed in a person-rem effective manner.

3.2 Internal Exposures

All individuals entering the reactor building will be monitored for internal radiation exposures according to established procedures. This monitoring will be accomplished by periodic whole body counting or bioassay, or both. All exposures to airborne radioactivity will be maintained ALARA and within the limits established in 10CFR20. Airborne radioactivity in work areas will be monitored according to established procedures. Air sampling for particulates will be performed using devices such as breathing zone air samplers and grab samples. Tritium grab samples will be taken as required according to established procedures.

An estimate of airborne radioactivity to be encountered by individuals performing plenum removal preparatory activities was derived from the personnel breathing zone air samples, for radioactive particulates, and tritium grab samples, for tritium, taken in the reactor building during the first quarter of 1984. These levels are given below.

	Reactor Building Concentration (μCi/cc)	Worker MPC (μCi/cc)	Fraction MPC with respirator * (MPC-hr/hr)
Cs-137	8.3E-8	1E-8	0.0083
Cs-134	3.9E-9	1E-8	0.0004
Sr-90	2.5E-9	1E-9	0.0025
H-3	1.0E-7	5E-6	0.020

* Protection factor for particulates = 1,000, for tritium = 1

The planned activities are not expected to increase the tritium or particulate levels inside the reactor building. The additional release of tritium to the reactor building atmosphere due to evaporation of the reactor coolant is not expected to increase the tritium level in the reactor building atmosphere because of the low tritium concentration in the reactor coolant of approximately 0.03 μCi/ml. The additional release of particulates to the reactor building atmosphere may result from water droplets falling off of retrieved tools and becoming airborne or the drying out of residue upon the inner diameter of the IIF. The speed of the upward air flow, associated with the evaporative process of the water, is expected to be small so that the water droplets, liberated from the tools, would fall back into the RV water rather than becoming airborne. The presence of water vapor in the air space above the water would minimize the drying out of any residue on the walls of the IIF and the associated particulate release.

3.3 ALARA Considerations

The objective of minimizing occupational exposure to levels ALARA has been a major goal in the planning and preparation for all activities in the containment. The actions that have been taken or are being planned toward meeting this objective are summarized in this section. These actions will minimize the time personnel must work in radiation fields, maximize the distance between personnel and radiation sources to the extent practicable, and utilize shielding where appropriate to meet the ALARA objective. Protective clothing and respirators will be used as necessary to reduce the potential for external contamination and internal exposure of personnel.

Execution of individual tasks are maintained ALARA by a detailed radiological review by Radiological Engineering and mock-up training. The need for the mock-up training will be determined on a case-by-case basis. The degree of difficulty and newness of the operation will influence the need for mock-up training and the detail of the mock-ups. This training will approximate the actual work situation as closely as can be achieved for each task utilizing appropriate equipment, protective clothing, and respiratory protection.

Inspection tooling has been designed with the intent of keeping radiation exposures ALARA. Counterbalances will be used to allow one operator to raise or lower tooling with no heavy exertion. To prevent tangling or severing of the cable or the draw wire the camera positioning draw wire and cable will be routed inside the tool handles. There are individual tool handles for each tool end to avoid interchanging tool handles inside containment. Tool assembly will be performed outside of containment except for attaching tool ends to tool handles and the final routing of the cable. One pin locks the tool handle to the tool end and the cable routing is an easy one-time occurrence. Flow holes exist in the tool ends to allow flooding of the hollow tooling so that buoyancy effects are minimized and radiation streaming is prevented. For ease of decontamination the tooling is fabricated of stainless steel and aluminum with smoothed inside and outside surfaces and no sharp edges or protrusions. Flushing and draining holes are provided with flushing capability from the top of the tooling.

Extensive planning of tasks to be conducted in a radiation field, and training of personnel will be used to reduce the time needed to complete a task. Extensive use of photographs and the in-containment closed circuit television system will be used to familiarize personnel with the work area. The higher radiation areas are identified to personnel and the work is structured to avoid these areas to the extent practical. Practice sessions will be utilized as necessary to ensure that personnel understand their assignments prior to entering the containment. Planning and training are proven methods of ensuring that personnel are properly prepared to conduct the assigned task expeditiously.

4.0 Safety Concerns

To ensure that the preparatory activities for plenum removal are conducted in a safe manner, safety concerns have been evaluated with respect to both plant safety and radioactive releases to the environment.

4.1 Plant Safety

To ensure that the plant remains in a safe condition all planned activities for this task, as scoped in Section 1.3, are evaluated with regard to maintaining the stabilization of the core. Planned inspection activities consist of video examination, measurements, and manipulations within the plenum assembly and the core cavity. Stabilization of the core is assured since the preventative measures to preclude a boron dilution event (see Section 2.0) are prerequisites for inspection activities.

Manipulations within the plenum assembly and the core cavity will be performed with extreme care and within given constraints because of their potential to disturb the core. The dislodging of fuel assembly end fittings will be performed provided that

1. core inspection reveals that the end fitting is unsupported (no full length rods exist) and
2. all unsupported end fittings from Batch 1 and Batch 2 fuel have been dislodged prior to dislodging any unsupported end fittings from Batch 3 fuel.

Implementation of these two operational constraints will ensure that the potential reactivity consequences will be negligible compared to worst case core configurations analyzed for a postulated head drop accident presented as an attachment to Reference 1. Core topography has shown, and video inspection will verify, that the APSRA's are unsupported and can therefore be driven into the core region with expectant core disruptions no more severe than the dislodging of unsupported end fittings. If the APSRA's cannot be driven into the core then attempts will be made to withdraw them into the plenum assembly provided that they are observed to have no full length rods, otherwise their vertical travel will be limited to five inches. The inadvertent dropping of tooling is expected not to disrupt the core to an extent greater than the dislodging of unsupported end fittings.

Since accessibility to the plenum is obtained by removing an IIF platform shield plate, the consequences of dropping a shield plate have been evaluated. The shield plates which vary in size ($\sim$ 6 square feet to $\sim$ 25 square feet) and in weight ($\sim$ 910 pounds to $\sim$ 3600 pounds assuming 1 inch of lead shielding) are removed one at a time. The removed plates are planned to be staged either on the IIF platform or in the refueling canal. The handling and moving of the plates will be in accordance with procedures or Unit Work Instructions (UWI's). If the plates are to be staged in the refueling canal, load paths 2 will be defined in the applicable procedure, UWI, or work package. These load paths will be administratively controlled to ensure that a postulated drop would not compromise plant safety or the structural integrity of the refueling canal floor.

The consequences of a postulated drop of an IIF shield plate above the reactor cavity impacting the IIF platform are evaluated below.

The IIF platform which totally covers the reactor vessel may collapse should a dropped IIF shield plate impact the IIF platform. The platform cover rests approximately 7.2 feet above the plenum cover plate. Assuming 1 inch of lead shielding, the IIF platform weighs approximately 19 tons. The total impact energy onto the plenum cover assuming a collapse of the IIF platform is conservatively determined to be approximately 276,000 foot-pounds. This is conservative because the majority of the weight is in the lead shielding which is located below the IIF platform cover (giving a shorter drop distance). In addition this collapse of the IIF platform, if taken mechanistically, would not impact the plenum with a single sudden impact but rather with a sequence of lower energy impacts which would result in a less severe total loading on the plenum. Reference 1 presents the analysis of a postulated drop of the reactor vessel head assembly onto the plenum. The impact energy afforded by the dropped reactor vessel head assembly has been calculated to be approximately 457,000 foot-pounds. The head drop bounds the IIF platform collapse for the following reasons:

1) the total impact energy due to collapse of the IIF platform is less than that due to the head drop, and

2) the impact area of the collapsed IIF platform is much greater than the point load drop considered in the head drop analysis resulting in a more distributed loading on the plenum.

The results of the postulated head drop analysis are summarized below.

1. With a boron concentration of 3500 ppm the core remains subcritical even for the worst case reconfiguration of the fuel.

2. The structural integrity of the reactor vessel and its support skirt is not compromised.

3. Resulting reactor vessel displacements do not cause stresses on attached piping to exceed their faulted condition stress limits given in Section III of the ASME Boiler and Pressure Vessel Code, 1974 Edition, precluding failure of attached piping.

4.2 Radioactivity Releases

The inadvertent dropping of tooling and the dislodging of fuel assembly end fittings are evaluated below with respect to the release of particulate and gaseous radioactivity. Any potential release of particulates would be effectively scrubbed by the water in the IIF. Even in the unlikely event of particulates becoming airborne, negligible particulate activity would be released to the environment since the containment atmosphere is exhausted via the containment purge system through its associated HEPA filtration system. The gaseous radioactivity assumed still present within the core (mostly Kr-85) may be securely trapped in the grain boundaries of fuel pellets or in the intact fuel rods. It is highly unlikely that the Kr-85 remaining in the core could be released from the dropping of tooling or the dislodging of a fuel assembly end fitting. However, the possible core disruption from the postulated collapse of the IIF platform could result in the release of the remaining Kr-85 core inventory. The analysis of releasing the Kr-85 core inventory is presented in Reference 1. The analysis showed a release of 37,400 curies of Kr-85 with a resulting whole body dose of 12 millirem and a skin dose of 980 millirem at the site boundary.

4.3 10CFR50.59 Evaluation

10CFR50, Paragraph 50.59, permits the holder of an operating license to make changes to the facility or perform a test or experiment, provided the change, test, or experiment is determined not to be an unreviewed safety question and does not involve a modification of the plant technical specifications.

A proposed change involves an unreviewed safety question if:

a) The possibility of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or

b) The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or

c) The margin of safety, as defined in the basis for any technical specification, is reduced.

The planned activities will not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated. The planned activities will not create the possibility of an accident or malfunction of a different type than any evaluated previously. Sections 4.1 and 4.2 have demonstrated that the consequences of postulated accidents associated with the planned activities are bounded by the postulated accidents associated with the removal of the reactor vessel head which has been shown not to be an unreviewed safety question. Since the operation of systems and equipment are in accordance with approved procedures to ensure compliance to technical specifications, the tasks included in this SER will not reduce the margin of safety as defined in the basis for any technical specification.

Therefore, it is concluded that the preparatory activities for plenum removal do not involve any unreviewed safety question as defined in 10CFR Part 50, Paragraph 50.59.

5.0 Conclusions

The planned activities and the tooling employed have been described and evaluated. The evaluations have shown that the planned activities and the tooling design follow the continued commitment to maintain radiation exposure levels ALARA. The performance of these activities is expected to reduce the radiation exposure associated with the plenum removal by identifying potential removal problems so that remedial actions can be planned in advance. It has also been shown that no detectable increase of radioactivity release to the environment results from the planned activities. The consequences of potential core disturbances and accidental releases of radioactivity have been shown to be bounded by the analyses presented in Reference 1 which concluded that the removal of the reactor vessel head can be accomplished without undue risk to the health and safety of the public. Therefore, this same conclusion is justified for the preparatory activities for plenum removal.

6.0 References

1. Safety Evaluation Report for Removal of the TMI-2 Reactor Vessel Head, Revision 5, February 1984

2. Safety Evaluation Report for the Operation of the IIF Processing System, Revision 1, May, 1984

3. GPUN Letter 4410-83-L-0059, dated March 30, 1983, regarding Man-Rem Estimate.