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4410-85-L-0100
Document ID 0241A

May 1, 1985

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

Dear Dr. Snyder:

Three Mile Island Nuclear Station, Unit 2 (TMI-2)
Operating License No. DPR-73
Docket No. 50-320
Plenum Removal Safety Evaluation Report - Revision 3

Attached for your use in review and approval of the proposed plenum removal activities is Revision 3 to the Plenum Removal Safety Evaluation Report (SER). Revision 3 deletes the lifting arm spacer plate from the plenum lift rigging, deletes the plenum assembly contamination barrier, updates the load drop section of the SER, and incorporates changes to those sections relevant to canal leakage collection and RCS processing. This revision supercedes all previous revisions of the subject SER.

Sincerely,

Robert E Rogan
for F. R. Standerfer
Vice President/Director, TMI-2

FRS/RBS/eml

Attachment

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

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GPU Nuclear

SER 15737-2-G07-106 REV. 3

ISSUE DATE April , 1985

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TMI-2

DIVISION

SAFETY EVALUATION REPORT

FOR

Plenum Lift and Transfer

COG ENG C. L. Rid DATE 1/10/85

RTR [signature] DATE 1/10/85

COG ENG MGR. C. L. Rid for R. L. Rider DATE 1/10/85

NO	DATE	REVISIONS	BY	CHECKED	GROUP SUPERVISOR	MGR DESIGN ENGINEERING
3	4/26/85	Issued For Use	[initials]	[initials]	ETS	KR for RLR
2	4/10/85	Issued For Use	[initials]	[initials]	ETS	AR for RLR
1	1/25/85	Issued For Use	[initials]	[initials]	ETS	[initials] for RLR
0	1/10/85	Issued For Use	[initials]	[initials]	ETS	KR for RLR
				CHIEF ENGINEER		

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PDR ADOCK 05000320
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EPD-33811 10/84

Rev.	SUMMARY OF CHANGE
0	Initial Issue for Use
1	Revised and reissued for use
2	Revised to delete lifting arm assembly spacer plate and update information for Load Drops Over Reactor Vessel
3	Revised to reflect changes in the methods for collecting leakage in the fuel transfer canal and RCS processing post plenum lift. Revised to delete the plenum assembly contamination barrier. Revised to clarify plenum assembly drop in fuel transfer canal.

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

BECHTEL

DOCUMENT TITLE: Safety Evaluation Report for Plenum Lift and Transfer

JOB 15737

~~SPEC. NO.~~ 2-G07-106

REV. 3

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1.0 Introduction

1.1 Background

The removal of the plenum assembly (PA) from the reactor vessel (RV) is necessary to gain access to the core region for defueling. The severe conditions within the RV caused by the March 28, 1979 accident necessitated the performance of deliberate preparatory activities to ensure a successful PA removal from the RV. These preparatory activities have been addressed in previous safety evaluation reports (SER's). Reference 1 addressed the inspection and cleaning of potential interference points which may restrict free movement of the PA and the dislodging of unsupported fuel assembly end fittings. Reference 2 addressed the initial lift of the PA by use of hydraulic jacks, the additional inspection and cleaning of the PA, and dislodging of fuel assembly end fittings required to prepare the PA for its removal from the RV.

1.2 Purpose

The purpose of this SER is to describe the activities associated with the removal of the PA from the RV and to evaluate these activities to assure that these activities can be performed while maintaining radiation exposures to personnel as low as is reasonably achievable (ALARA) and that the tasks can be accomplished without presenting undue risk to the health and safety of the public.

1.3 Scope

This SER covers the activities associated with the removal, transfer in air, and storage of the PA. The sequencing of these activities follows the initial lift by jacking of the PA addressed in Reference 2.

1.4 Organization

Section 2.0 consists of the descriptions of activities associated with the removal of the PA. The activities addressed include the installation of a dam, flooding of the deep end of the fuel transfer canal (FTC), removal of the internals indexing fixture (IIF) platform, installation of the defueling platform support structure, and the actual lifting and transfer of the PA. |1

Section 3.0 consists of the radiological considerations associated with the planned activities including an assessment of the expected external occupational radiation exposure and a presentation of the measures to be taken to maintain the occupational exposure ALARA. |1

Section 4.0 addresses the safety concerns associated with the planned activities including an evaluation with respect to 10CFR50.59.

Section 5.0 presents the conclusions of this SER.

2.0 Description of Activities

Prior to the lift and transfer of the PA from the reactor vessel various supportive activities must first be accomplished. Section 2.1 describes these supportive activities to prepare for the PA lift and transfer. Section 2.2 describes the actual PA lift and transfer operation and the tooling to perform the lift.

2.1 Supportive Activities

The transfer of the PA to the deep end of the FTC requires the filling of the deep end of the canal with borated water to a level that would provide adequate radiation shielding for the PA. A dam weighing approximately two tons will be fabricated from stainless steel plates and shapes and placed into position in the existing keyways along the east and west walls of the FTC on elevation 322'-6" to allow the deep end of the canal to be filled to a level above the floor of the shallow end of the canal. The five ton hoist attached to the main hook of the polar crane is planned to be used to install the dam. The main structure of the dam is a 3/8" x 6'-0" x 20'-10.25" stainless steel plate with stiffeners behind and along the perimeter of the plate. The dam will be installed within the keyway on each side of the FTC and made watertight by the installation of two redundant parallel inflatable gaskets along the dam edges which contact the canal walls and floor. The gaskets will be leak-tested to demonstrate zero leakage by filling the gap between the gaskets with water. Any leakage which may occur as a result of gasket deterioration will be collected in the new fuel pit in the shallow end of the FTC. The capability to remove water from the new fuel pit will be provided. 3

Following the installation and leak-testing of the dam gaskets, the flanges on the fuel transfer tubes will be removed and the deep end of the FTC will be filled with RCS grade borated water such that the water level will be approximstely at the 327'-8" elevation after the PA is in place. The deep end of the FTC will be filled with reactor coolant grade borated water using the Fuel Transfer Canal Fill System. The FTC level will be monitored by a bubbler with level indication and alarm in the control room. It is currently planned that fuel pool 'A' will be filled later. In the period between the filling of the FTC and fuel pool 'A' provisions will be made for preventing any leakage into the fuel pool through the transfer tubes.

The IIF platform located on top of the IIF will be removed and staged on the 347'-6" elevation. The removal of the IIF platform is required not only to allow for PA removal but also to allow for the installation of the steel support structure for the defueling work platform (DWP). The DWP support structure has four vertical columns that rest on the canal floor at elevation 322'-6" outside of the canal seal plate. The canal seal plate protective cover will be removed prior to the installation of these columns. The horizontal beams of this support structure will accommodate decking that can be used as a working platform. A soft flexible cover will be available to be installed over the IIF following removal of the IIF platform.

This cover is center supported to facilitate shedding water and would be used to prevent the accidental introduction of unborated water into the RCS.

The removal of the IIF platform necessitates the disconnection of the instrument lines for IIF processing which results in the loss of one of the three available methods to monitor reactor coolant system (RCS) water level. However, the two methods of RCS water level indication still available from the decay heat letdown line external to the reactor building and the tygon tube standpipe connected to the RCS 2A cold leg inside the reactor building provide adequate level indication.

The IIF processing and the RCS sampling pumps will be removed prior to the PA lift. Prior to plenum removal RCS water can be processed via the submerged demineralizer system (SDS) should the need arise. Sampling of the RCS will continue to be done via the RCS sampling pump which will be replaced after PA removal. Following removal of the plenum assembly the RCS water will be processed by the defueling water cleanup system (DWCS) when that system becomes operational. In the interim prior to the availability of the DWCS, RCS processing can be accomplished by letdown through the letdown lines to the reactor coolant bleed tank (RCBT), processing with SDS and make-up to the RCS through the normal make-up lines or through SDS via the IIF processing pump which may be reinstalled following PA removal. 3 3

2.2 Plenum Lift and Transfer

The 55 ton PA will be lifted using the polar crane and the tripod which have been used in the successful removal of the reactor vessel head. Attached to the tripod will be three pendant assemblies each having a lifting arm assembly attached to the bottom end to latch onto the underside of the plenum ribs. The ribs will be used as analyses show the normal lifting lugs may not be capable of supporting the plenum due to the transients to which they were exposed. The PA will be lifted free of the reactor vessel and transferred in air north to the deep end of the FTC. Removal of the PA from the reactor vessel will result in a water level decrease in the reactor vessel of up to 11 inches as the PA displaces approximately 1700 gallons of water (following PA removal the water level will be returned to its normal level by the make-up system). The PA will then be lowered onto the storage stand with the jacks still attached. 3

2.2.1 Lifting Equipment Design

The lifting equipment consists of one fixed length pendant assembly and two variable length pendant assemblies attached to the normal plenum assembly lift rig. The fixed length pendant will be positioned vertically above a point 15° from the PA Y axis towards the Z axis. The variable length pendants each contain a load positioner and are located 120° from each other and the fixed pendant assembly.

The pendant assemblies are approximately 17 feet long with the load positioners extended 6 inches. The load positioners will indicate load, load direction (up and down), and the amount of extension. Each positioner will have 12 inches of travel and be capable of re-positioning (raising or lowering the load $\pm$ 6 inches) the load from a 6 inch extension. The load positioners will be capable of remote operation with remote load/direction indication.

The lower end of each pendant assembly will have attached a lifting arm assembly. The lifting arm assembly consists of a lifting arm, lock plate assembly, cable pin assembly and a transfer block. The load is carried through the transfer block and lifting arm into the bottom side of one of the ribs in the cover of the PA. Also, the lifting arm, when in the loaded position, rests on the top of the adjacent rib which eliminates the rotational twist on the rib being lifted. The lifting arm is designed to lock under the plenum ribs and will be verified locked before lifting the plenum. The design of the lifting arm assemblies preclude disengagement while under load and are capable of remote disengagement once they are unloaded.

The pendant assemblies, including the load positioners, each have a design rating of 25 tons and are designed in accordance with NUREG-0612. Individual subassemblies will be load tested at 150% rated load, and have factors of safety of 3 to yield stress and 5 to ultimate stress, per ANSI N14.6. Non-load bearing members will be designed to meet allowable AISC stresses.

2.2.2 Travel Path

Prior to the PA lift by the polar crane the PA will have been elevated approximately 7.25" by use of the hydraulic jacks and be supported by the mechanical followers integral to the hydraulic jacks as described in Reference 2. A free lift path will be confirmed by inspection prior to lifting the PA.

The PA will be lifted clear of the reactor vessel, IIF and DWP support structure in air. Following completion of the vertical lift the PA will be moved approximately 32 feet to the north, submerged in the deep end of the FTC and then lowered onto the plenum storage stand. 3

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 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 continuously monitor dose rates in the reactor building during the plenum assembly lift and supportive activities.

The collective personnel radiation exposure to workers during the lift and transfer activities of the PA and during the supportive in-containment activities has been estimated. The estimate was developed based on projected person-hour requirements and reactor building exposure rates associated with these activities. The collective dose is estimated to be 30 person-rem. This figure is based on 500 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 accummulated by other groups in containment. Based on this, the estimate for radiological controls support is 6 person-rem, and the total for all groups is estimated at 36 person-rem.

Due to the uncertainty in the person-hour estimate and the radiological conditions which will exist during the inspection, lift and transfer activities, it is estimated that the total exposure could vary by up to $\pm$ 30 percent. Considering these uncertainties, 25 to 50 person-rem has been selected to be used as the estimate for the performance of the activities scoped in this SER, including radiological controls support.

The increase in radiation dose field inside the reactor building afforded by the dry lift and transfer of the PA is not expected to increase this estimated personnel exposure. This is due to a minimum number of personnel being inside the reactor building during the lift and transfer operation and those personnel will control and monitor the lift and transfer operation from within the lead curtain shielded area above the pressurizer missile shields. A direct line-of-sight path between the PA and in-containment personnel is planned only when the PA is submerged in water. Estimates based on conservative calculations have been made of the dose rates in air as a function of distance from the side of the PA. Measurements have shown that the actual dose rates are less than those calculated. However, the dose rates presented here for contingency planning are those based on the conservative calculations:

Distance from PA (feet)	Dose Rate from PA (R/hr)
3	120
6	80
10	45
20	16
40	4.8
70	1.7
100	0.86

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.

Respiratory protection has been used to minimize the uptake and deposition of airborne radioactivity in the body. The use of respiratory protection devices can, by reducing uptakes of radioactive materials, result in overall dose savings (internal and external); however, if they impede work, total dose can increase by causing an elevated external dose. Current radiation protection guidance as expressed in International Commission on Radiation Protection-26 (ICRP-26) considers both external and internal sources of dose and recommends minimizing the sum of them.

For soluble cesium-137, the internal dose is 2.5 mrem (received over several years, T_{eff} 70 days) for each hour of exposure at MPC. For soluble strontium/yttrium-90, the bone dose is approximately 15 mrem (received over 50 years, T_{eff} 6400 days) for each hour of exposure at MPC. Even if there is no overall savings in the total dose due to elimination of a respirator for a given task (that is, the increased internal dose exactly offsets decreased external dose), the fact that the internal dose is calculated on a fifty year dose commitment whereas external dose is deposited instantly means that the rate of dose deposition is reduced on an overall basis.

The Radiological Controls Department, via the prework radiological review process, shall determine if the use of respiratory devices for a task is ALARA. This review will examine the current radiological conditions in the work area, the potential of the task or other concurrent tasks to perturb the radiological conditions and when available, review the results of previous airborne activity measurements in the work area for similar tasks.

An estimate of airborne radioactivity to be encountered by individuals performing the initial lift activities was derived from the personnel breathing zone air samples, for radioactive particulates, and tritium grab samples taken in the reactor building prior to, during, and following reactor vessel head removal. These levels are given below.

Isotope	Reactor Building Concentration (μCi/cc)
Cs-137	2.5E-9
Cs-134	6.8E-11
Sr-90	2.3E-11
H-3	1.1E-7

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 the PA and potentially "drying-out". Because of the short time duration that the PA will not be in water the amount of particulates potentially becoming airborne is not expected to increase the particulate concentration in the reactor building atmosphere.

3.3 ALARA Considerations

The objective of minimizing occupational exposure 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. A mock-up simulating the 3 point lift and transfer including a simulated plenum cover with appropriate lift ribs, a simulated internals handling fixture (tripod) and the actual pendant assemblies will be used. Extensive training of workers on the mock-up will familiarize the workers with tasks to be performed. This training will result in less in-containment time and personnel exposure.

Equipment has been designed with the intent of keeping radiation exposures ALARA by minimizing in-containment assembly, simplifying operation, and having remote operation capability. The lifting arm assemblies attached to the bottom of the pendant assemblies are self-latching and once unloaded can be detached remotely. The load positioners can be remotely operated from approximately 200 feet.

The extensive use of video equipment to monitor the lift and the levelness of the lift allows workers not to be in the "line of sight" of the PA while it is being transferred from the reactor vessel to the deep end of the FTC.

Savings of in-containment time is also realized by simplifying the communication and control required during the lifting operations by having a central area for monitoring and control. This central area is the same shielded area located above the pressurizer missile shields on the 370'-4" elevation used during the reactor vessel head lift operations.

4.0 Safety Concerns

To ensure that the lift and transfer of the PA and the supportive activities 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 are evaluated with regard to maintaining the stability of the core. Stability of the core could be compromised by either:

1) a dilution of the boron concentration in the RCS that causes the boron concentration to fall below the concentration limit required to maintain the core in a subcritical condition, or

2) a loss of RCS water that would uncover the core.

Section 4.1.1 addresses boron dilution concerns and the procedural requirements to preclude a boron dilution event. Section 4.1.2 addresses the evaluation of postulated drops of loads carried over the reactor vessel, with respect to the potential of causing a loss of RCS inventory which could uncover the core. Section 4.1.3 addresses the evaluation of a postulated drop of the PA in the shallow portion (El. 322'-6") of the FTC. A postulated drop of the PA in the deep end (El. 308'-0") of the FTC would not impact plant safety as the consequences of such a drop would not affect the stability of the core, drain or reduce the water level in the RCS or decrease containment accessibility.

4.1.1 Boron Dilution

The RCS is presently borated to a level of approximately 5050 ppm boron. Reference 3 has shown that a boron level of 4350 ppm will assure that the core would remain subcritical at any core configuration and this minimum concentration will be maintained during and following PA removal. Procedures governing the frequency of boron sampling, RCS level monitoring, isolation barrier checking, and boron dilution source checking will be in force during the various plant

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operations (e.g., termination of IIF processing, IIF platform removal, and canal fill) to ensure against a boron dilution event that would cause the boron concentration in the RCS to drop below 4350 ppm.

4.1.2 Load Drops Over Reactor Vessel

4.1.2.1 Prior to IIF Platform Removal

All load handling activities to be performed over the IIF platform (prior to its removal from the IIF) will be evaluated against the criteria that follow:

An analysis of postulated load drops onto the Internals Indexing Fixture (IIF) platform with the Plenum Assembly supported on jacks has been performed. This analysis examined several categories with resulting load weight vs. maximum lift height limitations (note, lift heights are measured from the top of the IIF platform to the bottom-most portion of the suspended load).

Loads that will be handled above the IIF platform have been broken down into four categories. The first two categories envelope the handling of objects that could fit through a 24" x 48" opening and weigh up to 5500 pounds. Category 3 includes loads up to 5500 pounds that cannot fit through a 24" x 48" opening. Category 4 includes the handling of objects less than or equal to 1800 pounds that may be handled over an open slot and will fit through the opening. 2

The guidelines provided for the 4 categories identified, shall be used during removal and replacement of the platform cover plates for access to the plenum area.

Any number of plates may be removed at one time as long as they are handled in accordance with category 4 loads when over open slots. Plates in excess of 1800 pounds in weight such as the triangular plates should be handled as those in category 3.

Plates being relocated on the IIF platform may be placed only on plates which are installed in the platform. Plates may not be stacked more than two high, including the installed plate.

1. Objects which when rotated in any orientation can fit through a 24" x 48" opening.

Weight (lbs)	Maximum Lift Height
500	17'-0"
1000	5'-0"
1500	2'-9"
2000	1'-10"
2500	1'-4"
3000	1'-0"

Compliance with the maximum lift heights will ensure that in the event the load is dropped neither the IIF platform nor any of the individual platform plates will collapse. The 24" x 48" opening size represents the largest single opening in the IIF platform within the perimeter of the IIF cylinder.

2. Objects greater than or equal to 10'-0" in length and that can pass through a 24" x 48" slot if turned on end.

Weight (lbs)	Maximum Lift Height
3000 to 5500	1'-0"

2

Loads less than 3000 lbs can be treated under #1 above, those greater than 5500 lbs are excluded. Loads in this group must be rigged horizontally. Compliance with these guidelines will ensure that in the event the load is dropped, neither the IIF platform or any of the individual platform plates will collapse.

3. Objects which when rotated in any orientation will not fit through a 24" x 48" opening.

Weight (lbs)	Maximum Lift Height
500	59'-0"
1000	17'-0"
1500	8'-6"
2000	5'-5"
2500	3'-10"
3000	3'-0"
4000	2'-0"
5000	1'-6"
5500	1'-0"

Compliance with these guidelines will ensure that the IIF platform does not collapse. However, it is possible that, due to the configuration of objects in this category, an IIF platform plate may be deformed sufficiently to allow it to be driven through its own slot onto the plenum.

An IIF platform plate (maximum weight 1800 lbs for the largest plate which can be driven through its own slot) that is pushed through the IIF platform would fall through the water in the IIF and onto the plenum assembly. Because of the shape of the plate, even after it is deformed, the water will reduce the velocity at which the plate impacts the plenum assembly. In addition, the plate is likely to strike the guide tubes that are above the top of the plenum assembly. As a result of the above, the impact load transmitted to the plenum assembly will be considerably reduced. The 55 ton plenum assembly is currently supported approximately 7 1/2" above its "normal" or rest position on 4 jacks. These jacks are each rated at 50 tons and transfer all loads to the C.S.A. and R.V. flange. In as much as the postulated impact energy will be minimal, it is considered extremely unlikely that the jacks would fail or that any significant impact energy would be transferred to the lower grid assembly. On this basis, it is concluded that an IIF platform plate falling onto the plenum assembly will not develop sufficient impact energy to result in the breaking of an incore instrument guide tube.

An analysis of a postulated drop of the reactor vessel head assembly onto the plenum was presented in Reference 5. The analysis demonstrated that with a total impact energy onto the plenum assembly of approximately 457,000 foot-pounds, the structural integrity of the reactor vessel and its support skirt is not compromised. In addition, resulting reactor vessel displacements do not cause stresses on attached piping to exceed their faulted stress limits given in Section III of the ASME Boiler and Pressure Vessel Code, 1974 Edition, precluding failure of attached piping. In comparison, the drop of an 1800 pound plate onto the jacked plenum would impart less than 10,000 foot-pounds or approximately 2% of the energy of the postulated head drop. 3

4. Objects less than or equal to 1800 pounds that may be handled over an open slot and can fit through the open slot.

Weight (lbs)	Maximum Lift Height
200 to 1800	1'-0"
less than 200	4'-0"

Loads to be handled over an open slot are limited to the largest single platform cover plate which could be postulated to fall through the open slot during removal, or tools which will be operated from the IIF platform level. The maximum lift height for the platform cover plate during removal shall be restricted to 1'-0". This height is adequate for removal. The maximum lift height for tools shall be limited to 4'-0". This height is adequate to allow lifting the tool over any temporary safety rails placed over the open slot.

The evaluation for objects in this group is enveloped by the 1800 lb. plate which can be driven through its own slot as presented for group 3. Due to the configuration of the triangular plates, they cannot be dropped or driven through their own slots and strike the plenum.

Maximum allowable load lift heights will then be defined in the applicable procedure or unit work instruction (UWI) for the handling of the load.

4.1.2.2 Following IIF Platform Removal

The handling of heavy loads over the reactor vessel following removal of the IIF platform will be governed by the "Safety Evaluation Report for Heavy Load Handling over the TMI-2 Reactor Vessel", Reference 6. Reference 6 is the controlling document for all heavy load handling activities occurring above or in the reactor vessel including plenum lift. The Referenced SER demonstrates that a drop into the vessel of any heavy load up to the limits described in the SER will not impact the health and safety of the public.

4.1.3 Plenum Drop in the FTC

A postulated drop of the PA has also been evaluated to determine if such a drop onto the 322'-6" elevation could cause a rupture of the incore instrumentation guide tubes routed within the incore instrumentation cable chase on elevation 282'-6". An assessment has been made of the resulting point impact loading on the 322'-6" floor afforded by a 12 foot drop of the PA onto the floor. This assessment determined that due to the floor thickness and geometry the impact load would be transferred to the primary and secondary shield walls primarily in shear. Local concrete spalling at the underside of the floor was judged not to occur, and even if minor spalling were to occur there would be no damage to the in-core tubes due to the following: 3

1) The location of the floor and walls with respect to the in-core instrument trench will preclude a direct hit by debris generated due to local spalling and,
2) the heavy bottom reinforcement (#11 reinforcing bars at 6" centers each way) provided.

Based on the results of this analysis, it is concluded that a drop of the PA in the shallow end of the FTC would not cause any damage to the in-core instrument guide tubes.

4.1.4 Polar Crane Failure

A mechanical failure of the polar crane or its rigging may result in a plenum assembly drop. A mechanical or electrical failure could result in the PA being suspended in air above the reactor vessel or fuel transfer canal. Any of these failures could create a radiation hazard for operations personnel (see Section 3.1 for the expected dose rates). If recovery from these failures requires shielding of the plenum assembly, this may be provided by filling the fuel transfer canal to normal refueling level with borated water.

Sections 4.1.2 and 4.1.3 of this SER demonstrate that the postulated worst case plenum assembly drops will not uncover the fuel in the reactor vessel or cause criticality. None of these postulated polar crane failures will significantly increase airborne activity levels in containment above normal recovery levels or impact containment integrity. Consequently, polar crane failures will not present undue risk to the health and safety of the public.

4.2 Radioactivity Releases

The planned activities associated with the lift and transfer of the PA are not expected to release any appreciable amounts of gaseous or particulate activity. Any potential gaseous radioactivity would be the assumed Kr-85 activity still present in the reactor core which may be securely trapped in the grain boundaries of fuel pellets or in the intact fuel rods. The release of Kr-85 from the movement of the PA is considered highly unlikely. Even in the unlikely event of particulates becoming airborne during PA movement, negligible particulate activity would be released to the environment since the containment atmosphere would be exhausted via the containment purge system through its associated HEPA filtration system.

The postulated load drops over the reactor vessel could impart a loading on the vessel that may cause the release of some of the remaining Kr-85 inventory in the core. A bounding analysis is presented in Reference 5 which assumes the total release of the remaining Kr-85 core inventory. The amount released is assumed to be 37,400 curies of Kr-85 with resulting doses estimated to be 12 millirem to the whole body and 980 millirem to the skin, for an individual located at the nearest site boundary. These doses are well within the acceptance criterion given in NUREG 0612, "Control of Heavy Loads at Nuclear Power Plants."

4.3 10 CFR50.59 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. The handling of loads over the reactor vessel will be subject to lift height limitations such that the consequences of postulated load drops above the reactor vessel would be bounded by previously evaluated accidents. The drop of the PA has been shown to be extremely unlikely. However, the unimpeded fall of the PA from its maximum lift height to its normal position in the reactor vessel has been analysed and the analysis is described in Reference 6. This analysis demonstrates that this drop of the PA falls within the bounds of the analysis performed in Reference 5. Since the operation of systems and equipment are in accordence 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. 3

Therefore, it is concluded that the lift and transfer of the PA does not involve any unreviewed safety question as defined in 10CFR Part 50, Paragraph 50.59.

5.0 Conclusions

The PA lift and transfer, and associated activities, have been described and evaluated. The evaluations have shown that the task and the tooling employed follow the continued commitment to maintain radiation exposure levels ALARA. The evaluations have also shown that no detectable increase of radioactivity releases to the environment will result from the planned activitea. The consequences of postulated accidents with respect to potential core disturbances and loadings on the reactor vessel have been shown not to compromise plant safety. The accidental releases of radioactivity have been evaluated and are bounded by the analyses presented in Reference 5. It is therefore concluded that the lift and transfer of the PA, and the associated activites, can be performed without presenting undue risk to the health and safety of the public.

6.0 References

1. "Safety Evaluation Report for Preparatory Activities for Plenum Assembly Removal," GPU letter 4410-84-L-0140 dated August 28, 1984, F. R. Standefer to B. J. Snyder.

2. "Safety Evaluation Report for Initial Plenum Lift", GPU letter 4410-84-L-0164 dated November 2, 1984, F. R. Standefer to B. J. Snyder.

3. "Criticality Report for the Reactor Coolant System at TMI-2," GPU letter 4410-84-L-0199 dated November 8, 1984, F. R. Standefer to B. J. Snyder.

4. "Load Drops Prior to Plenum Removal," GPU letter 4410-84-L-0212 dated November 21, 1984, F. R. Standefer to B. J. Snyder.

5. "Safety Evaluation Report for Removal of the TMI-2 Reactor Vessel Head," GPU letter 4410-84-L-0014 dated March 9, 1984, B. K. Kanga to B. J. Snyder.

6. "Safety Evaluation Report for Heavy Load Handling Over the TMI-2 Reactor Vessel" 15737-2-G07-110. 2