Nuclear

GPU Nuclear Corporation
Post Office Box 480
Route 441 South
Middletown, Pennsylvania 17057-0191
717 944-7621
TELEX 84-2386
Writer's Direct Dial Number:

(717) 948-8461

4410-84-L-0164
Document ID 0083A

November 2, 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
Initial Plenum Lift Safety Evaluation Report

Attached for your review and approval is the Safety Evaluation Report (SER) for initial plenum lifting. This SER covers the period beginning with the completion of the inspection activities through jacking of the plenum to a height of nine (9) inches. Various activities (i.e., additional cleaning, inspections, and measurements) will be performed during plenum jacking.

This SER concludes that initial lifting of the plenum does not constitute an Unreviewed Safety Question and can be performed without undue risk to the health and safety of the public.

Pursuant to the requirements of 10 CFR 170, an application fee of $150.00 is attached.

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

Sincerely,

F R Standerfer

F. R. Standerfer
Vice President/Director, TMI-2

D009 w/check
1/1 $150.00
00015194

FRS/RBS/jep

Attachment (GPU Nuclear Check No. 00013194)

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

SAFETY EVALUATION REPORT
FOR
INITIAL PLENUM LIFT

THREE MILE ISLAND UNIT 2

Rev.salon 1
October, 1984

TABLE OF CONTENTS

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. Ordinarily, the PA can be lifted out of the reactor vessel (RV) immediately following the removal of the RV head. However, the severe conditions caused by the March 28, 1979 incident necessitates additional preparatory activities. These additional preparatory activities include inspection and possible cleaning of potential interference points which may restrict free movement of the PA and the dislodging of unsupported fuel assembly end fittings. These activites have been described in Reference 1. Upon completion of these preparatory activites the PA will be lifted up to a height of 9" by the use of hydraulic jacks to ensure a controlled initial lift without risking a potential sudden release of the spring energy in the polar crane and its cables. During the initial lift various hold points will allow additional inspections and cleaning including the removal of fuel assembly end fittings suspended from the PA.

1.2 Purpose

The purpose of this Safety Evaluation Report (SER) is to describe the activities associated with the initial lift up to a height of 9" of the PA, 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, and that the performance of these activities can be done while maintaining radiation exposures as low as is reasonably achievable (ALARA).

1.3 Scope

This SER covers the activites associated with the initial lift of the PA. The sequencing of these activites is after the preparatory activities necessary for this initial lift as scoped by Reference 1 and prior to the final lift, transfer, and storage of the PA.

1.4 Organization

Section 2.0 consists of the descriptions of the activites associated with, and supportive of the initial lift. The tooling and equipment employed for these activities are also described.

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 ALARA.

Section 4.0 addresses the safety concerns associated with the planned activites 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

2.1 Preparatory Activites

Prior to the initial lift of the PA the following describe the in-containment conditions.

- o The IIF is filled to approximately 5 feet with water having a boron concentration satisfying Reactor Coolant System (RCS) technical specification and operating procedural requirements.
- o The auxiliary fuel handling bridge is available to be used as a working platform.
- o Provisions are in place to isolate containment (if required to isolate).
- o A crane is available for use.
- 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 is available in the reactor building via portable power distribution centers.

In addition to the conditions given above, actions specified in Reference 2 will be incorporated in procedural requirements to be in effect to monitor RCS boron content and to preclude inadvertent boron dilution of the RCS during the initial PA lift activities.

2.2 Initial Lift Activities

The PA shall be in a condition to allow for its final lift out of the reactor vessel and its transfer to its storage location in the refueling canal, following completion of the initial lift activities. Thus, the initial lift activities shall include, not only the lifting of the PA, but also the dislodging of fuel assembly end fittings and any final cleaning of the PA required. Hold points during the initial lift have been instituted to allow for these supportive activities.

The initial lift shall be performed using four hydraulic jack assemblies which satisfy the stress requirements and safety factors of ANSI B30.1-1981, "Jacks."

Each jack assembly consists of a hydraulic jack cylinder, a special mainframe structure, a mechanical follower and two displacement transducers. The mechanical follower is incorporated into each jack to provide mechanical support for the plenum assembly when the hydraulic jack pressure is relieved. Such hydraulic depressurization is planned during various lift hold point activities, such as end fitting separation, to minimize the potential for leakage of the hydraulic fluid from this high pressure system. During normal operation, the air motor will continuously hold the mechanical follower against the hydraulic cylinder, advancing as the hydraulic cylinder is extended. Thus, as the hydraulic pressure to the jack is relieved, the load of the plenum assembly will be transferred to the mechanical follower. The operation of each lifting jack assembly is accurate to within ± 0.015 in.

The jack assemblies will be installed on the plenum cover in the triangular area formed by the intersecting plenum ribs at each of the four reactor vessel axes. The jack assemblies extend down into the 2 3/4" annulus between the plenum cylinder top flange and the core support shield. The jacks lift up under the plenum ribs and push down on the top of the core support shield flange.

The actual lifting operation will be conducted by personnel located on the 347 foot elevation. The equipment will be arranged so that one individual can monitor the various jack assembly instruments while supervising each of the operators of the four independent hydraulic hand pumps. Hydraulic, air, and electrical lines connect the jacks to a control station located at the south end of the refueling canal on the 347 foot elevation. The control station consists of the necessary remote control equipment for the initial lift. The control cabinet contains individual jack readouts for jack displacement, mechanical follower position, hydraulic pressure, and air pressure. The cabinet also contains the on/off and forward/reverse valves for controlling the air motors on the mechanical followers. Adjacent to the cabinet are the individual, manually operated hydraulic pumps that control the lift force and displacement of the jack assemblies. The video equipment will be on the auxiliary fuel handling bridge which will be viewable and accessible from the jacking locations.

The primary PA-to-core support assembly interfaces which may present potential interferences are listed below according to the plenum lift height at which they are encountered.

Elevation (in)	Interfaces
0	Vessel Key to Plenum Keyway.
	26 Pairs of LOCA Restraint Bosses.
	Core Support Shield (CSS) Bottom Flange to Plenum Upper Grid Ring Forging (UGRF).
	End Fittings to End Fittings.
	End Fittings to Grid Pads.
	CRGA/APSRA to C-Tubes.
1/2	End Fittings Begin to Separate from Grid Pads.
	End Fittings and CRGA Separate.
3	CSS Bottom Flange/UGRF Begin to Separate.

	LOCA Bosses Separate.
7	CSA LOCA Bosses Begin to Align with Upper Grid Ring Forging.

2.3 Supportive Activities

Supportive activities for the initial lift are the handling of loads over the reactor vessel, the dislodging of fuel assembly end fittings, and the cleaning of the plenum. The handling of loads over the reactor vessel includes the removal and staging of the IIF cover shield plates and the placement of the jack assemblies. All load handling will be in accordance with procedures or Unit Work Instructions (UWI's). The dislodging of fuel assembly end fittings will be performed at various hold points during the initial lift. Any final cleaning of the PA required will be performed at the 9" hold point in preparation for PA removal. The tooling used to perform the end fitting separation and cleaning is described in Reference 1.

The video monitoring system for the initial lift jacking operation will utilize video cameras to monitor the initial lift of the plenum assembly. Two cameras will be used to observe and monitor that the upper grid ring forging does not separate from the upper grid rib section. These cameras will monitor this joint at two symmetrical locations 180 degrees apart. They will be inserted between the plenum assembly cover plate and the inside of the plenum cylinder and lowered to the proper elevation for monitoring separation between these components. Other cameras equipped with rotatable right angle lens may be inserted into the core cavity via control rod guide tubes. These cameras will be used to monitor the condition of the fuel assemblies on the periphery of the core cavity, if required. A camera will also be used to monitor the level indicating device located on the plenum cover plate.

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 initial lift activities of the plenum assembly.

The collective personnel radiation exposure to workers during the initial lift activities has been estimated. The estimate was developed based on projected person-hour requirements and reactor building exposure rates associated with these initial lift activities. The person-rem estimate of 21 person-rem is based on 350 in-containment person-hours. | 1

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 4 person-rem, and the total for all groups is estimated at 25 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, 18 to 33 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. 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 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 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 apace 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 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 detailed mock-up, the Interface Test Assembly (ITA), simulating the configuration and orientation of the upper portions of the plenum cover assembly and the core support shield on which the jack assemblies rest, located in the Defueling Test Assembly (DTA), will be utilized. The triangular jack seating areas located at the plenum cover assembly horizontal component axes, and enclosed by the plenum upper structural ribs and the plenum support ring will be representative of actual tolerances on the ITA. Extensive training of workers on the DTA/ITA and other mock-ups with the jacks will familiarize the workers with the tasks to be performed. This training will result in less in-containment time and personnel exposure.

Tooling has been designed with the intent of keeping radiation exposures ALARA. Tool assembly will be performed outside of containment except for attaching tool ends to tool handles and the final routing of camera 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.

Savings of in-containment time is realized by simplifying the communication and control required during the initial lifting sequences by having a centralized Operating and Control Station. This centralized station allows all operations to be conducted with constant monitoring and continuous line-of-sight control. To minimize radiation exposure to personnel the Operating and Control Station is positioned in an area where the radiation level is low and there is easy accessibility to and from the area. Savings of in-containment time and radiation exposure is also realized by simplifying to the extent possible the operation of the jacks. The installation of the jacks, or any required removal for maintenance, is expected to be relatively rapid and easy with no complicated special tooling required. To prevent having to use complex rigging schemes to lower and position the jacks, the jack hoist rings are adjustable which will allow a jack to be balanced prior to taking it into the containment building.

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 initial lift of the PA is 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 activites for this task are evaluated with regard to maintaining the stability of the core. Planned activities include the initial lift of the PA to 9", the dislodging of remaining fuel assembly end fittings, and the final inspection and any cleaning of the PA required in preparation for PA removal. Stability of the core is assured since the preventative measures (see Section 2.1) are prerequisites to initial lift activitea.

The initial lift with the associated dislodging of fuel assembly end fittings and plenum cleaning have a potential to disturb the core. The analysis presented in Reference 3 shows that the core will remain subcritical at any core configuration provided that the boron level in the RCS remains above the limit specified in the reference. Reference 2 provides the actions that will be performed to preclude a boron dilution. The total quantity of hydraulic fluid present in the hydraulic jacking system is approximately eight gallons. Thus a postulated total loss of the hydraulic fluid into the RCS would not significantly dilute the boron concentration in the RCS and would not impact the subcriticality of the core.

The design of the jack assemblies which incudes a mechanical follower to provide mechanical support for the PA during the hold points and final lift height of 9" precludes the plenum from dropping. The mechanical follower advances as the hydraulic cylinder is extended via an air-operated motor. Upon loss of air or malfunction of the advancing mechanism further advancement of the follower would be prohibited. Each jack assembly has a jack displacement transducer (JDT) and a mechanical displacement transducer (MDT) with readouts at the jacking control station. The JDT measures the displacement of the jacking foot and the MDT measures the extension of the mechanical follower. These transducers allow for continuous monitoring of the jack displacement and mechanical follower extension. The jack assemblies have each been designed and tested for a 60-ton static load. Based on the "as-found" plenum dimensions at the jack locations, the limiting stress conditions result in rating the jacks at 50 tons. 1

Postulated load drops over the reactor vessel prior to the jacking of the PA have been evaluated in Reference 1. Reference 1 has shown that the impact energy onto the PA from the postulated total collapse of the IIF platform is less than the impact energy onto the PA from the postulated point load drop of the reactor vessel head presented in Reference 4. The collapse of the IIF platform would bound any load handling mishap which could occur over the reactor vessel and thus was chosen as the "worst-case" load drop over the reactor vessel. The consequences of this "worst-case" load drop have been shown in Reference 1 to be less than the consequences of the point load drop of the reactor vessel head which have been shown to be acceptable in Reference 4.

Loads carried over the reactor vessel at jacking hold points and after the PA has been jacked to its 9" lift height shall be subject to additional constraints to preclude jack failure from a postulated load drop impacting the PA. The PA may be impacted by a dropped load which falls through an opening in the IIF platform when shield plates are removed for end-fitting separation, PA inspection, or PA cleaning. The PA may also be impacted by portions of the IIF platform which fall into the reactor vessel from potential structural failure of the IIF platform caused by dropped loads impacting the IIF platform. Loads handled over the reactor vessel will be evaluated on a case by case basis to determine maximum lift heights over the reactor vessel. These maximum lift heights will be defined in applicable procedures or UWI's written for the handling of these loads.

4.2 Radioactivity Releases

The planned activities associated with the initial lift of the plenum 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 dislodging of fuel assembly end fittings or the movement of the PA is considered highly unlikely. 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 postulated collapse of the IIF platform 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 4 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.

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. The hydraulic jacks used to perform the initial lift of the PA have been designed with mechanical followers to ensure that the PA can be safely held in its elevated position. 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 postulated accidents associated with the removal of the reactor vessel head (which has been shown not to be an unreviewed safety question) bound the consequences of postulated load drops above the reactor vessel. 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 initial lift of the plenum does not involve any unreviewed safety question as defined in 10CFR Part 50, Paragraph 50.59.

5.0 Conclusions

The initial lift of the plenum assembly is necessary to ensure that the plenum can be removed from the reactor vessel. The severe conditions resulting from the March 28, 1979 incident may restrict plenum movement because of distortion and/or debris accumulation at various interfaces between the plenum and the core support shield. Jacking the plenum provides a controlled initial lift, an allowance for dislodging of remaining fuel assembly end fittings, final cleaning and inspection of the plenum, and assurance that the plenum can be lifted and removed from the reactor vessel.

The intial lift and associated activites 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 activites. 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 4. It is therefore concluded that the initial lift of the plenum assembly 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," Rev. 2, June, 1984

2. "Hazards Analysis: Potential for Boron Dilution of Reactor Coolant System," document 4430-84-007R

3. "Criticality Report for the Reactor Coolant System at TMI-2," document 15737-2-N09-001, October 1984

4. "Safety Evaluation Report for Removal of the TMI-2 Reactor Vessel Head", Rev. 5, February, 1984