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April 19, 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
Safety Evaluation Report for Heavy Load Handling
Over the TMI-2 Reactor Vessel

Attached for your review and approval is the Safety Evaluation Report (SER) for handling heavy loads over the TMI-2 Reactor Vessel. This SER considers the data obtained from the recent video inspections of the lower head of the reactor vessel. The SER concludes that heavy load handling activities can be performed over the TMI-2 Reactor Vessel without undue risk to the health and safety of the public.

Per the requirements of 10 CFR 170, an application fee of $150.00 is enclosed for review of this document.

Sincerely,

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

FRS/RBS/eml

A033
1/1 w/check
$150.00

Enclosure: GPU Nuclear Check No. 00015519

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

TRANS. NO. DEDT-1501

TRANSMITTAL

FROM: R. L. Rider
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TO: GPU NUCLEAR TMI-2 DIVISION
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ATTN: Supervisor, Records Services - A. Stowe

DATE: April 18, 1985

BECHTEL JOB 15737
PURCHASE ORDER NO.
BECHTEL FILE NO. 0741/G07-110

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ACTION NUMBER

1. Approved. Manufacturing may proceed.
2. Approved. Submit final dwg. Mfg. may proceed.
3. Approved except as noted. Make changes and submit final dwg. Mfg. may proceed as approved.
4. Not Approved. Correct and resubmit.
5. Review not required. Mfg. may proceed.
6. Comment/Approval Due Date ____
7. Information Only
8. Other Issued For Use

DESCRIPTION

Action No.	Bechtel Number	Manufacturer's Number	Rev.	Title
8	15737-2-G07-110	-	0	Safety Evaluation Report for Heavy Load Handling Over the TMI-2 Reactor Vessel

292
RFR/tlj

(See Attached Sheet for Distribution List)

Very truly yours,

C. L. R.
fo R. L. Rider
Manager, Design Engineering

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SER 15737-2-G07-110 REV. 0

ISSUE DATE April 18, 1985

☒ ITS

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TMI-2

DIVISION

SAFETY EVALUATION REPORT

FOR

HEAVY LOAD HANDLING

OVER THE TMI-2 REACTOR VESSEL

COG ENG C. L. Ril DATE 4/10/85

RTR E.T. Smith DATE 4/18/85

COG ENG MGR. C L Ril for RLR DATE 4/18/85

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NO	DATE	REVISIONS	BY	CHECKED	GROUP SUPERVISOR	MGR DESIGN ENGINEERING
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EPD-33611 10/84

Rev.	SUMMARY OF CHANGE

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: Heavy Load Handling over the TMI-2 Reactor Vessel

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SPEC. NO. 2-G07-110	0
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1.0 Purpose, Scope and Organization

1.1 Purpose

The purpose of this Safety Evaluation Report (SER) is to demonstrate that heavy load handling activities directly above and within the reactor vessel through to the completion of reactor fuel removal can be accomplished without presenting undue risk to the health and safety of the public.

1.2 Scope

The scope of this SER includes the handling of heavy loads (loads greater than 2400 lbs. including rigging weight) in the vicinity of the reactor vessel. This includes any load handling activity which could result in a heavy load drop onto or into the vessel either directly or indirectly (by causing the collapse of structures/equipment installed over the vessel). This SER addresses all such load handling activities through to the completion of reactor vessel fuel removal activities but excludes removal of the core support assembly.

This SER addresses the potential impact of heavy load handling activities on the integrity of the reactor coolant system; it does not address the potential damage to the item dropped or the consequences of that damage (e.g. damage to a dropped defueling canister and the consequences of canister damage are not addressed in this SER). This SER also addresses reactivity control and radiological consequences of a heavy load drop into the reactor vessel.

Heavy load handling activities outside the area over the reactor vessel are addressed in the "Safety Evaluation Report for Heavy Load Handling Inside Containment" (Reference 6) and the "Safety Evaluation Report for Plenum Lift and Transfer" (Reference 9).

1.3 Organization

Section 2.0 of this SER describes the range of load handling activities over and within the reactor vessel anticipated during the recovery program. Section 3.0 addresses the potential safety concerns associated with these load handling activities and the features provided to prevent and/or mitigate any potential impact on the health and safety of the public. Section 4.0 addresses the radiological considerations and Section 5.0 addresses each of the four NUREG-0612 criteria for satisfactory control of heavy loads. Section 6.0 addresses the 10 CFR50.59 criteria for determining whether these load handling activities involve an unreviewed safety question. Sections 7.0 and 8.0 present the conclusions of this safety evaluation and a list of references, respectively.

2.0 Description of Activities and Equipment

2.1 Prerequisites for Heavy Load Handling over the Vessel

The following is a list of prerequisites for performing any heavy load handling activity addressed in this SER.

(i) The performance of load handling activities will be by qualified personnel trained in the operation and safety of lifting and handling equipment

(ii) An approved procedure or Unit Work Instruction applicable to the activity is available

(iii) The crane, lifting rigging and attachment points shall have been inspected and tested in accordance with approved procedures

2.2 Heavy Load Handling Activities

During recovery it will be necessary to lift heavy loads over and within the open reactor vessel to support inspection and defueling of the vessel. The maximum load currently identified is that of the plenum and its rigging (approximately 73 tons).

Other than the plenum, the loads carried over the vessel will include support structures, shielding and equipment associated with the recovery program which must be installed around or over the reactor vessel. Loads handled within the vessel include defueling canisters, defueling equipment and fuel debris. Lift heights will be restricted in accordance with the guidelines presented in Section 3.1.2.1 of this safety evaluation report.

As a result of physical constraints within the containment or other considerations including appropriate ALARA practices it may be necessary or prudent to transfer some loads over the vessel en route to another area of the containment, typically the deep end of the fuel transfer canal. This load path over the reactor vessel will only be used after an assessment of alternative load travel paths.

Heavy load handling activities will be performed and supervised by personnel trained and qualified in accordance with the "TMI-2 Lifting and Handling Program," 4000-PLN-3891.01 (Reference 4). For heavy lifts over the vessel, a lift supervisor will be assigned control of the lift and must certify that all prerequisites of the operation are satisfied. The lift supervisor will have no other duties other than to supervise the lift operation.

Crane operators are trained and qualified in accordance with Reference 4. This includes classroom instruction in the use of the load handling cranes, written or oral examinations to demonstrate knowledge of the equipment to be operated, and performance of practical proficiency tests. Reference 4 also requires operators to meet the qualification requirements of ANSI B30.2.

Lifts over the vessel will be rigged in accordance with an approved rigging sketch which will identify the type and size of each component in the lift rig.

The majority of heavy load lifts over the reactor vessel will be performed only once prior to the completion of reactor vessel fuel removal. The only identified loads which will be handled on a frequent basis are the defueling canisters and the shield plugs for the defueling work platform tool slot and miscellaneous equipment items.

There are approximately 230 defueling canisters which will be handled over the vessel. When empty the canisters weigh less than 2400 pounds and are therefore not heavy loads. When filled with water and fuel debris the maximum canister weight is approximately 3350 pounds in air. When the canisters are in the filled condition they will be lifted out of the vessel and transferred to the deep end of the fuel transfer canal by a crane (Canister Handling Bridge) specifically designed to prevent a drop of the canister. The crane "hook" is an air operated grapple which provides positive engagement, verified by an indicator light, between the crane and the canister. After the canister is brought to the fully raised position within its transfer shield, redundant retention mechanisms are engaged to prevent the canister from dropping in the event of grapple failure or accidental release. This crane has a quality classification of Important to Safety for all components whose failure could result in dropping a canister.

The components and structure of the Canister Handling Bridge (CHB) are designed to conform to safety standards of ANSI B30.2 and ANSI B30.16. Prior to the use in defueling, the CHB will be tested to meet the requirements of ANSI B30.2 and the TMI-2 Lifting and Handling Program. The critical load bearing components of the CHB trolley and transfer shield, including the canister retention mechanisms, were designed to conform to NUREG-0612, Section 5.1, i.e., the factor of safety for materials used is 5 based on ultimate strength and 3 based on yield strength.

Some of the defueling work platform shield plugs weigh in excess of 2400 pounds. These plugs will be lifted out of their positions in the work platform whenever access is required beneath the plugs for fuel removal activities, filter canister removal, etc. All of the plugs weigh less than 5 tons and will normally be handled by the Reactor Building Service Crane (RBSC). The RBSC is designed and will be tested in accordance with CMAA Specification No. 70 for Electric Overhead Traveling Cranes, Class C (Moderate Service) and ANSI B30.2, Overhead and Gantry Cranes.

2.3 Load Handling Equipment

Heavy load handling equipment, including cranes, hoists, slings and hooks are designed and constructed in accordance with the applicable ANSI B30 series standards with the exception of the reactor building polar crane structural components which were designed and constructed in accordance with CMAA #70. The rated capacity of the equipment is certified by operational and proof tests. Special load handling devices will be designed, constructed and tested in accordance with ANSI N14.6-1978. In addition the stress design factors stated in ANSI N14.6 will be based on the combined static and dynamic loading as specified in NUREG-0612, Section 5.1.1(4).

The lifting of the 55 ton plenum assembly will be accomplished with the use of the polar crane, tripod, and the pendant assemblies described in Reference 9. The polar crane and tripod have been used in the successful removal of the 170 ton reactor vessel head and have also lifted approximately 214 tons for the polar crane load test. Each pendant assembly is load tested to 150% of its design rating of 25 tons and has, by design, safety factors of 3 to yield stress and 5 to ultimate stress.

Heavy load handling equipment is inspected in accordance with Reference 4. This reference also describes the testing program for initial certification of equipment and re-certification of equipment which has been modified.

Preventive maintenance is performed on load handling equipment in accordance with the manufacturer's recommendations at one year intervals or less.

The integrated program of conservative equipment design, control of equipment fabrication, inspections, tests, and maintenance minimize the potential for load handling equipment failure.

3.0 Safety Concerns

3.1 Load Drop Accidents

Loads dropped over or within the reactor vessel have the potential to indirectly or directly damage the incore instrument tubes which penetrate the vessel lower head. Since these instrument tubes are part of the reactor coolant system boundary, damage to the tubes or to the penetration of the tube at the lower head could result in loss of water from the reactor coolant system (RCS). The exact nature of the damage and the resulting reduction in reactor vessel water level are difficult to determine, but if the loss of water can be shown to be within the makeup capability to the vessel, the core will remain covered during postulated drop accidents. No other credible RCS failure modes could lower the water level below the elevation of the bottom of the reactor vessel outlet nozzles (plant elevation 314'-0").

3.1.1 Prevention of a Load Drop Accident

The potential for a load drop accident into the reactor vessel is minimized by careful control of load handling activities and the use of load handling equipment which has been conservatively designed and tested as described in Section 2.0.

A drop of the plenum assembly by failure of the polar crane or the tripod is extremely unlikely since the weight of the plenum assembly and attendant rigging and attachments is approximately 73 tons which is less than one-half of the current rating of the polar crane and is less than one-sixth of the crane's original design capacity rating of 500 tons. A drop of the plenum assembly by failure of the pendants is also extremely unlikely based on their factors of safety.

Load handling activities are performed in accordance with approved procedures for such activities including 4000-PLN-3891.01 "TMI-2 Lifting and Handling Program" (Reference 4). As noted in Section 2.2, loads are not carried over the reactor vessel without first evaluating alternative load paths.

Each specific load handling activity is controlled by a Unit Work Instruction or procedure which is reviewed by a Responsible Technical Reviewer in accordance with TMI-2 procedures. Load lifting and handling activities will be performed by personnel who have been trained and qualified for these activities as described in Section 2.2.

3.1.2 Reactor Vessel Integrity

3.1.2.1 General

The only postulated failure mechanism which could potentially lower the water level in the reactor vessel below the bottom of the coolant pipe nozzles is damage to the incore instrument tubes at the point where they penetrate the lower vessel head. There are a total of 52 such penetrations distributed in the lower head, as shown in Figure 3 of Reference 3, and each configured generally in accordance with Figure 3-1 of this safety evaluation report (the penetration detail shown is for the center penetration). Figure 3-2 shows the overall configuration of the lower core support assembly, four of the 52 incore instrument guide tubes and nozzles, and the lower vessel head.

It should be noted that at present there is no measurable leakage from the reactor coolant system. Some loads have already been exerted on the top of the core debris bed during recovery activities performed to date (e.g. sample probes and dropped partial fuel assemblies during plenum jacking and end fitting removal) without causing leakage from the vessel. In addition, as noted in Reference 8 reactor lower head video inspections performed thus far do not show damage to the visible incore instrument nozzles and the lower vessel head. Nor do they show evidence of bridging by debris buildup between the visible incore instrument guide tubes and incore instrument nozzles.

In addition, from March 1979 until removal of the vessel head in July 1984, the RCS was subjected to various internal pressures including > 1,000 psig immediately following the accident, 300 $\pm$ 60 psig from May 1979 to April 1980, 90 $\pm$ 10 psig from April 1980 to April 1984 and 50 psig until June 15, 1984 when depressurization occurred.

Reference 2 presents an analysis of a postulated drop of the reactor vessel head onto the vessel flange and onto the plenum assembly. This analysis shows that the structural integrity of the reactor vessel and its support skirt are not compromised and the resulting reactor vessel displacements do not cause stresses on the attached piping, including the incore instrument tubes, to exceed their faulted condition stress limits given in Section III of the ASME Boiler and Pressure Vessel Code, 1974 Edition, thus precluding failure of attached piping.

A separate analysis has been performed for a drop of the plenum assembly (PA) 7.5 feet through air and an additional 14.8 feet in water. The analyzed PA load weight is 73 tons, which includes PA, rigging, jacks, etc. For conservatism, this analysis assumes the PA would fall unimpeded through the Internals Indexing Fixture, though this is not considered credible due to the clearances involved and the existence of indexing keys.

The analysis considers the bouyancy and frictional drag forces afforded by the water. The resulting kinetic energy of the PA at impact is approximately 1.2 million foot-pounds. The kinetic energy of the dropped reactor vessel head at impact is approximately 1.7 million foot-pounds, or 42% greater than the kinetic energy of the falling PA.

Based on analysis, the maximum allowable lift height of the PA is 22.3 feet. However, it is expected that the actual lift height will be well below this height.

The lift heights for loads other than the plenum assembly to be handled over the reactor vessel will be restricted to ensure that in the event the load is dropped, the impact energy transmitted to the reactor vessel or any internal reactor vessel components will be less than or equal to the impact energy resulting from a reactor vessel head drop as analyzed in Reference 2. The lift height and weight restrictions will differ depending on whether the IIF platform, plenum assembly, and defueling work platform are installed or removed. The restrictions are presented as four cases which cover all currently identified configurations through to the end of vessel defueling.

Case 1 - Plenum Assembly In Vessel on Jacks and IIF Platform Installed

When the plenum assembly (PA) is supported by jacks and the IIF platform installed, any load that would be dropped over the reactor vessel is assumed to first impact the IIF platform causing it to collapse onto the raised PA. Further, it is assumed that the dropped load and collapsed IIF platform cause the jacks to fail and the PA to fall to its pre-raised position. The total impact energy transmitted to the core support assembly from the IIF platform and the PA has been calculated to be approximately 330,255 foot-pounds. This would leave approximately 126,745 foot-pounds of impact energy available for the dropped load in order to remain within the bounds of the reactor vessel head drop analysis. The following expression conservatively defines the allowable lift height, H (feet), for a load of weight, W (pounds):

$$H = \frac{126{,}745}{W} + 322.5$$

Where H is the maximum plant elevation to which the load may be raised and W is the weight of the lifted load including the weight of rigging which is rigidly attached to the load, in pounds.

Case 2 - Plenum Assembly In Vessel On Jacks and IIF Platform Removed

When the plenum assembly (PA) is supported by jacks and the IIF platform is removed it is assumed that a dropped load causes the jacks to fail and the PA to fall to its pre-raised position. The total impact energy transmitted to the core support assembly from the PA has been calculated to be approximately 58,000 foot-pounds. This would leave approximately 399,000 foot-pounds of impact energy available for the dropped load in order to remain within the bounds of the reactor vessel head drop analysis. The following expression conservatively defines the allowable lift height, H (feet), for a load of weight, W (pounds):

$$H = \frac{399,000}{W} + 322.5$$

Where H is the maximum plant elevation to which the load may be raised in feet, and W is the weight of the lifted load including the weight of rigging which is rigidly attached to the load, in pounds.

Case 3 - Plenum Assembly Removed and Defueling Work Platform Not Installed

When the plenum assembly has been removed from the reactor vessel and the defueling work platform has not yet been installed, the following expression conservatively defines the allowable lift height, H (feet), for a load of weight, W (pounds):

$$H = \frac{457,000}{W} + 322.5$$

Where H is the maximum plant elevation to which the load may be raised in feet, and W is the weight of the lifted load including the weight of rigging which is rigidly attached to the load, in pounds. In addition, no load will be raised above plant elevation 405'-0".

Case 4 - Plenum Assembly Removed and Defueling Work Platform Installed

An analysis was performed to calculate load handling limits for load handling over the defueling work platform (DWP). This analysis examined the entire platform to determine the weakest point and impact loads were applied at that point. The load and lift height limits calculated in this analysis preclude the collapse of the DWP. The following table presents the results of the analysis:

Weight (lbs)	Allowable Plant Elevation
2000	340.3
5000	334.7
10000	333.1
20000	332.3
30000	332.0
40000	331.9
50000	331.8

3.1.2.2 Instrument Tube Integrity Outside Vessel

The incore instrument tubes outside of the vessel are 1/2" Schedule 80, 304L stainless steel pipes, each welded to a 3/4" schedule 160 Inconel 600 nozzle penetrating the vessel wall. Reference 3 presents an evaluation which demonstrates that the integrity of the incore instrument tubes outside of the reactor vessel was not adversely affected by the March 28, 1979 accident or by the environment (both internal and external) to which the tubes have been exposed since the accident. Based on the load drop analyses described in Section 3.1.2.1 and the evaluation presented in Reference 3, a failure of incore tubes outside of the reactor vessel as a consequence of a load drop onto the vessel flange or plenum assembly for the maximum load to be handled over the vessel is not considered credible.

3.1.2.3 Instrument Tube Integrity Inside Vessel

As shown in Figure 3-1 each incore instrument tube terminates inside the vessel at the incore instrument nozzle. Above each incore instrument nozzle and separated by a vertical gap of several inches is the incore instrument guide tube which is attached to the lower core support assembly. The only physical connection between the incore instrument nozzle and the incore instrument guide tube is the incore instrument detector assembly which consists of a cluster of nine detectors within an inconel sheath with a wall thickness of 0.021 inches.

The entire core support assembly is supported from a ledge in the interior wall of the reactor vessel just below the reactor vessel flange (See Figure 3-2). A heavy load drop into the vessel onto the plenum assembly or onto the core debris bed following removal of the plenum (and including a drop of the plenum back into the vessel) can cause a downward deflection of the lower core support assembly. The impact energy of the single heaviest load to be handled over the vessel, i.e. the plenum, will be less than the impact energy of the head drop analysis presented in Reference 2. The resulting lower core support assembly deflection calculated for the head drop is approximately 1/2 inch, considerably less than the normal vertical clearance between the incore instrument nozzle and the incore instrument guide tube.

The downward deflection of the lower core support assembly can exert a force on the incore instrument nozzles by one or more of the following three mechanisms:

a. Transmission of force by the incore instrument detector assembly
b. Bridging of the gap between the guide tubes and the nozzles by fuel debris
c. Guide tube impact directly onto the nozzles

Since the incore instrument assembly is contained within a 0.021 inch sheath, the cables and sheath would buckle before they could transmit a damaging force to the nozzles.

Although it is possible that fuel debris bridging may have occurred between the nozzles and the guide tubes, it is highly unlikely due to the umbrella configuration as shown in Figure 3-1.

If local deflection of the lower core support assembly due to a point load drop onto the debris bed after the removal of the plenum exceeds the vertical clearance between the guide tubes and the nozzles, the nozzles could be damaged. The nozzle is shown in phantom in Figure 3-1 at the point at which the nozzle will seat within the guide tube and the guide tube will exert its full downward force on the nozzle. If the load impact is great enough to cause a gross failure of the lower core support assembly, the downward movement of the assembly may be restrained by the core guide lugs which are 0.833 inches below the lower grid shock pads when the lower core support assembly is in its normal position (See Figure 3-2). If the shock pads fail, most or all of the incore instrument nozzles may be damaged.

After the plenum is removed and during the early stages of defueling, the upper layer of the core debris bed will provide some protection for the incore tube nozzles by absorbing some of the load drop energy. This layer consists of gravel-like material which will transmit less energy to the lower core support assembly then would be transmitted by a solid debris bed.

3.1.2.4 Postulated Failure Mechanism

If a heavy load drop is postulated, and if it is further postulated that the resulting downward deflection of the lower core support assembly is sufficient to cause damage to the incore instrument nozzles, the worst anticipated failure mechanism is the shearing off of the nozzle at the inside vessel wall. As shown in Figure 3-1, the 3/4" schedule 160 portion of the instrument tube which penetrates the vessel wall is welded directly to the vessel wall. The incore instrument nozzle is welded separately to the vessel wall and the 3/4" pipe. Consequently, failure of the nozzle is unlikely to fail the 3/4" pipe to vessel weld which provides the penetration seal. For conservatism, however, it is assumed that this weld fails as a result of the postulated load drop accident.

Failure of the tube-to-vessel-wall weld will not result in the tubes being forced out of the lower head by the head of water in the vessel. The tubes consist of schedule 80 stainless steel pipe and are supported at the floor below the vessel. The maximum clearance, taking into account manufacturing tolerance, between the OD of the tube and the ID of the bore in the vessel wall is 0.010 inches. There is insufficient flexibility in the tubes to allow them to drop the 5 1/2 inches required to fall free of the bottom of the vessel head.

As noted previously, incore tube failure outside of the vessel is not considered credible. Consequently the only credible leakage path from the vessel following a heavy load drop is through the annulus around the tube penetrations through the vessel wall.

3.1.3 Mitigating Features

In the event of a heavy load drop into the reactor vessel, a RCS leak will be detected by one or more of the level monitoring systems provided for the RCS. Two level monitoring systems are installed outside of the reactor vessel (one connected to an RCS hot leg and the other to an RCS cold leg) and are therefore not subject to damage from a load drop accident in the vessel. Low RCS level alarms are provided in the control room.

An analysis of the maximum leakage rate possible through the incore tube lower head penetrations, assuming a constant water level of 327'-6" in the Internals Indexing Fixture and a 0.010 inch annular gap (See Figure 3-1), demonstrated that the resulting flow from failure of all 52 penetrations would be less than 20 gpm (less than 0.40 gpm per penetration). This analysis assumed the maximum static head of water in the vessel prior to the load drop accident and assumed the worst case manufacturing tolerances for all 52 penetrations. The analysis takes no credit for the tube-to-vessel weld at the inside wall or the local inconel buildup on the outside of the vessel.

In the unlikely event of a reactor vessel leak, the core debris bed will be maintained covered with borated water by means of makeup from the Borated Water Storage Tank (BWST) and recirculation from the basement via the Decay Heat Removal (DHR) system or a recirculation system to be installed for this purpose.

As described in Reference 5, the BWST, located outside the reactor building, has a capacity of about 460,000 gallons of makeup water for the RCS. It is the primary source of makeup water to the RCS in the event of a leak. It is planned to maintain a minimum of 390,000 gallons of water in the BWST in the future except as permitted in accordance with procedures approved pursuant to Technical Specification 6.8.2. However, prior to plenum lift the BWST inventory may be temporarily reduced to 310,000 gallons to flood the deep end of the fuel transfer canal. Consequently, for conservatism a 310,000 gallon BWST inventory is assumed.

Two methods are available to transfer the water from the BWST to the RCS; (1) by installed pumps, and (2) by gravity flow. If power is available and pumps are utilized, the entire quantity of water in the BWST is available for makeup. If it is assumed that AC power is not available, approximately 220,000 gallons of the 310,000 gallons minimum inventory in the BWST is available as makeup by gravity flow via the reactor core flood nozzle at Elevation 317'-6". Analyses show that substantially more flow capacity is available by gravity flow than the 20 gpm leak.

The water level in the Internals Indexing Fixture (IIF) during recovery operations is planned to be at Elevation 327'-6", but future operations could possibly lower it to 321'-6" (just below the reactor vessel flange) to facilitate fuel handling operations. The reactor water level will always be at 321'-6" elevation or higher prior to any postulated incore leak. Under these conditions, with an incore leak of 20 gpm it will take in excess of 14 hours for the water level to drop to the 314' elevation. This provides ample time to detect the leak and take action to initiate pumped or gravity flow from the BWST to the RCS even in the event of a postulated five hour loss of electrical power.

Once gravity makeup to the RCS is initiated, the approximately 220,000 gallons of water available in the BWST will provide sufficient water to makeup for the 20 gpm leak rate for approximately 7.6 days. Use of the pumps after restoration of off-site power would permit using the remaining 90,000 gal. of BWST water and would provide a total of about 10.7 days of water storage for RCS makeup.

Makeup to the reactor vessel can be provided by recirculating water from the reactor building basement via the DHR system. Each DHR system pump has a capacity of 3,000 gpm at the rated head and will be operated as required to maintain the required water level in the reactor vessel.

Prior to initiation of recirculation flow from the reactor building basement to the RCS, it will be necessary to sample the basement water and assure that it is borated to the minimum allowable boric acid concentration in the RCS. Since it is possible that the basement boron concentration will be less than this, methods of attaining the proper boron concentration have been investigated and would be implemented prior to initiation of recirculation to the RCS. If it is conservatively assumed that the reactor building basement contains 70,000 gallons of unborated water, there are two alternative methods available to assure that the basement is sufficiently borated prior to initiation of recirculation flow. These alternatives are described in Reference 5 which demonstrates that the boron concentration in the RCS will be maintained above 4350 ppm following a load drop into the vessel.

The TMI-2 project has initiated action to procure a portable, dedicated pumping system to provide recirculation to the RCS from the reactor building basement. The system, including a minimum of two submersible pumps and all required hoses and valves, will replace the DHR system as the means of recirculation following a load drop into the vessel and permit the removal of the DHR from the Technical Specifications in accordance with Reference 5. The system will be procured, demonstrated to be operable and placed in dedicated, on-site storage. In the event of a load drop into the vessel resulting in an RCS leak, the submersible pumps will be lowered into the reactor building basement (with hose(s) attached) and connected to an installed power receptacle. The discharge end of the hose(s) will be placed directly into the IIF. The complete system installation can be accomplished well within the period during which makeup is provided from the BWST.

3.2 Criticality

As described in References 1 and 7, the reactor vessel core debris bed will remain subcritical if the RCS is maintained at a boron concentration of 4350 ppm or greater. As noted above, following the loss of RCS water as a result of a heavy load drop accident, the subsequent operation of the makeup and recirculation systems will maintain the core debris covered and will not decrease the boron concentration in the RCS below 4350 ppm. Consequently, the core will remain subcritical following postulated load drop accidents.

In the event that the postulated load drop accident introduces into the core debris bed materials which could act as neutron moderators or could cause localized deboration, the potential for localized criticality events could exist. Prior to removal of the plenum assembly, loads that may be dropped into the reactor vessel will impact the top of the plenum and thus will not contact the core debris bed. In addition, after the installation of the defueling work platform, loads handled in accordance with the height and weight guidelines provided in Section 3.1.2.1 will not, if dropped, contact the debris bed if the tool slots in the platform are closed or if the load cannot fit through the tool slots in any orientation. Consequently, for these cases, no restrictions on materials will be required. For all other heavy load handling activities over the vessel to ensure that localized criticality will not occur, the materials of each load will be evaluated on a case by case basis prior to performing the load handling activities.

3.3 Release of Radioactivity

3.3.1 Normal Conditions

There are no identified heavy load handling activities over the reactor vessel which will increase the release of radioactivity from the site for non-accident conditions.

3.3.2 Accident Conditions

The only significant source of radioactivity available for release as a consequence of a load drop accident into the vessel is the Kr-85 activity assumed to still be present in the reactor core which may be securely trapped in the grain boundaries of fuel pellets or in the intact fuel rods. The load drop impact may cause the release of some of the remaining Kr-85 inventory. A bounding analysis has been performed which assumes an instantaneous total release of the unaccounted for Kr-85 inventory. The amount released is assumed to be 31,300 curies of Kr-85 with the resulting dose estimated to be 9.7 millirem to the whole body for an individual located at the nearest site boundary and 1.8 mrem to the whole body for an individual located at the Low Population Zone (LPZ) Boundary. The meteorological dispersion parameters (X/Q) used were 6.1 X 10^{-4} sec/m^3 at the site boundary and 1.1 X 10^{-4} sec/m^3 at the LPZ boundary (as indicated in the FSAR).

During the reactor building basement recirculation mode following the postulated load drop accident, some particulate radioactivity may become airborne from evaporation of water on the reactor building basement floor and evaporation of possible leakage to the auxiliary building from the DHR system. Particulate activity which becomes airborne, will be collected on the HEPA filters in the ventilation exhaust system serving the area or, in the event of filter failure, contained within the area by isolation of the ventilation systems. Consequently, this particulate activity will not contribute significantly to the offsite doses.

4.0 Radiological Considerations

Radiological considerations for heavy load handling over the reactor vessel are not significantly different than for other activities performed in the reactor building, excluding potential releases due to a load drop. Radiological considerations, such as occupational external and internal exposures and special radiological precautions will be addressed in documents generated for specific tasks, including procedures, unit work instructions and radiological review documentation.

5.0 NUREG-0612 Evaluation

This section evaluates the results of load drops postulated in this SER against the four criteria set forth in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants."

5.1 NUREG-0612 Criteria

5.1.1 Criterion I:

Releases of radioactive material that may result from damage to spent fuel based on calculations involving accidental dropping of a postulated heavy load produce doses that are well within 10 CFR Part 100 limits of 300 rem thyroid, 25 rem whole body (analyses should show that doses are equal to or less than 1/4 of Part 100 limits).

Response to Criterion I:

Any activity releases caused by the load drops addressed in this SER would be released within the containment. The containment would act as a physical barrier and prevent any liquid releases from escaping to the environment. Likewise, any additional particulates that may become airborne would be removed by the high efficiency particulate air (HEPA) filters so as not to exceed the limits established in Criterion I.

The analysis described in Section 3.3.2 (for Krypton 85) showed that even when utilizing "worst case" assumptions (instantaneous total release with no containment), the maximum whole body dose is 9.7 millirem compared to a limit of 6250 millirem.

5.1.2 Criterion II:

Damage to fuel and fuel storage racks based on calculations involving accidental dropping of a postulated heavy load does not result in a configuration of the fuel such that k_{eff} is larger than 0.95.

Response to Criterion II:

Reference 1 conservatively demonstrates that with the RCS maintained at a boron concentration of at least 4350 ppm, no re-configuration of the fuel debris in the reactor vessel, including a re-configuration resulting from a heavy load drop, can cause criticality. Reference 1 states that k_{eff} is less than 0.99 at 4350 ppm boron concentration and shows that k_{eff} should be less than 0.97 at 5000 ppm boron concentration. These are conservatively calculated k_{eff}; actual k_{eff} is much lower. Therefore, these values meet the intent of the criterion and are adequate for the TMI-2 recovery period. The RCS is normally maintained at a boron concentration of 5050 $\pm$ 100 ppm. In the event that a heavy load drop causes leakage from the vessel, the makeup and recirculation systems provided will not cause the RCS boron concentration to drop below 4350 as described in Section 3.1.3.

Consequently the load drops postulated in this SER cannot cause a criticality event.

5.1.3 Criterion III:

Damage to the reactor vessel or the spent fuel pool based on calculations of damage following accidental dropping of a postulated heavy load is limited so as not to result in water leakage that could uncover the fuel (makeup water provided to overcome leakage should be from a borated source of adequate concentration if the water being lost is borated).

Response to Criterion III:

As described in Section 3.1.3, sufficient makeup and recirculation capability is provided to ensure that in the event of the maximum postulated water leakage from the vessel, the fuel debris bed will remain covered and adequate boration will be maintained. Load handling activities which could potentially damage the integrity of the vessel cannot also damage the makeup flow path.

5.1.4 Criterion IV:

Damage to equipment in redundant or dual safe shutdown paths, based on calculations assuming the accidental dropping of a postulated heavy load, will be limited so as not to result in loss of required safe shutdown functions.

Response to Criterion IV:

The only required safe shutdown functions that apply to the TMI-2 reactor vessel in its current cooling mode and core configuration are:

1) the capability to maintain subcriticality,
2) decay heat removal, and
3) the capability to maintain the integrity of components whose failures could result in excessive off-site releases.

Reactor coolant will be maintained in the reactor system (RCS) above the reactor vessel nozzles for decay heat removal and reactivity control. Subcriticality will be maintained as described in section 5.1.2. Currently, decay heat is removed by heat losses to ambient which has been demonstrated adequate to remove all decay heat produced by the core material in the reactor vessel as long as the water level in the reactor vessel is a minimum of el. 314 feet (Reference 2). As noted in Section 3.1.3 the water level in the reactor vessel will be maintained above el. 314 feet. Therefore, the ability to adequately remove the decay heat by the losses to ambient mode of cooling will be maintained. As such, no additional equipment is necessary to remove decay heat. The offsite releases are discussed in Section 5.1.1 and are well within acceptable limits.

Consequently, safe shutdown will be maintained for load handling and load drop accidents postulated in this SER.

In addition to the safe shutdown functions, the RCS water level will be maintained to provide personnel shielding.

6.0 10CFR 50.59 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 probability 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 heavy load handling activities will not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated. The design features and administrative controls described in section 3.1.1 ensure that the probability of an accident is not increased. The makeup and recirculation capability as described in section 3.1.3 ensure that the consequences of an accident will not be increased.

The postulated types of accidents associated with heavy load handling over the reactor vessel are load drops into the reactor vessel which have been considered in the TMI-2 FSAR. Therefore, load handling over the reactor vessel will not create the possibility of an accident or malfunction of a different type than previously evaluated.

The bases for the technical specifications are to maintain the fuel in a subcritical condition and to maintain decay heat capability. As discussed in Section 3.1.3, the fuel will remain covered with water having a boron concentration of a minimum of 4350 ppm. Based on Reference 1, this will ensure that the fuel will remain subcritical. Also as discussed in Section 3.1.3, the water will be maintained at a minimum level of el. 314 feet, which based on Reference 2, will ensure continued decay heat removal capability.

Therefore it is concluded that heavy load handling activities over the reactor vessel do not involve any unreviewed safety questions as defined in 10CFR Part 50, Paragraph 50.59.

Reference 5 requests that the DHR system be deleted from the technical specifications and replaced with another recirculation system. However, until the replacement system is installed and operable the DHR system is available to provide the recirculation capability required. After Reference 5 is implemented, the proposed recirculation system will be available and included in the technical specifications. In addition, the technical specification changes relating to boron concentration described in Reference 10 are required to support the removal of the plenum assembly. No technical specification changes beyond those proposed in Reference 10 are required for handling heavy loads over the reactor vessel.

7.0 Conclusions

The descriptions and evaluations presented in this SER demonstrate that heavy load handling activities over the reactor vessel will be performed in a safe manner and are unlikely to result in a load drop accident. However, should a drop accident occur the evaluations demonstrate that any damage to safety related systems and equipment will be limited such that mitigating systems will maintain the reactor in a safe condition. The evaluations also demonstrate that any potential increase in radioactive releases as a consequence of a heavy load drop accident will be within allowable limits. It is therefore concluded that the heavy load handling activities described in this SER can be performed without presenting undue risk to the health and safety of the public.

8.0 References

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

2. "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.

3. "Safety Evaluation Report for Measuring Fuel Material Collected in the Lower Region of the TMI-2 Reactor Vessel via Gamma Profiling of the In-Core Detectors", GPU letter 4410-85-L-0002 dated January 3, 1985, F. R. Standerfer to B. J. Snyder.

4. "TMI-2 Lifting and Handling Program", Number 4000-PLN-3891.01, Revision 0.

5. Technical Specification Change Request No. 46, GPU letter 4410-84-L-0154, dated November 6, 1984, F. R. Standerfer to B. J. Snyder.

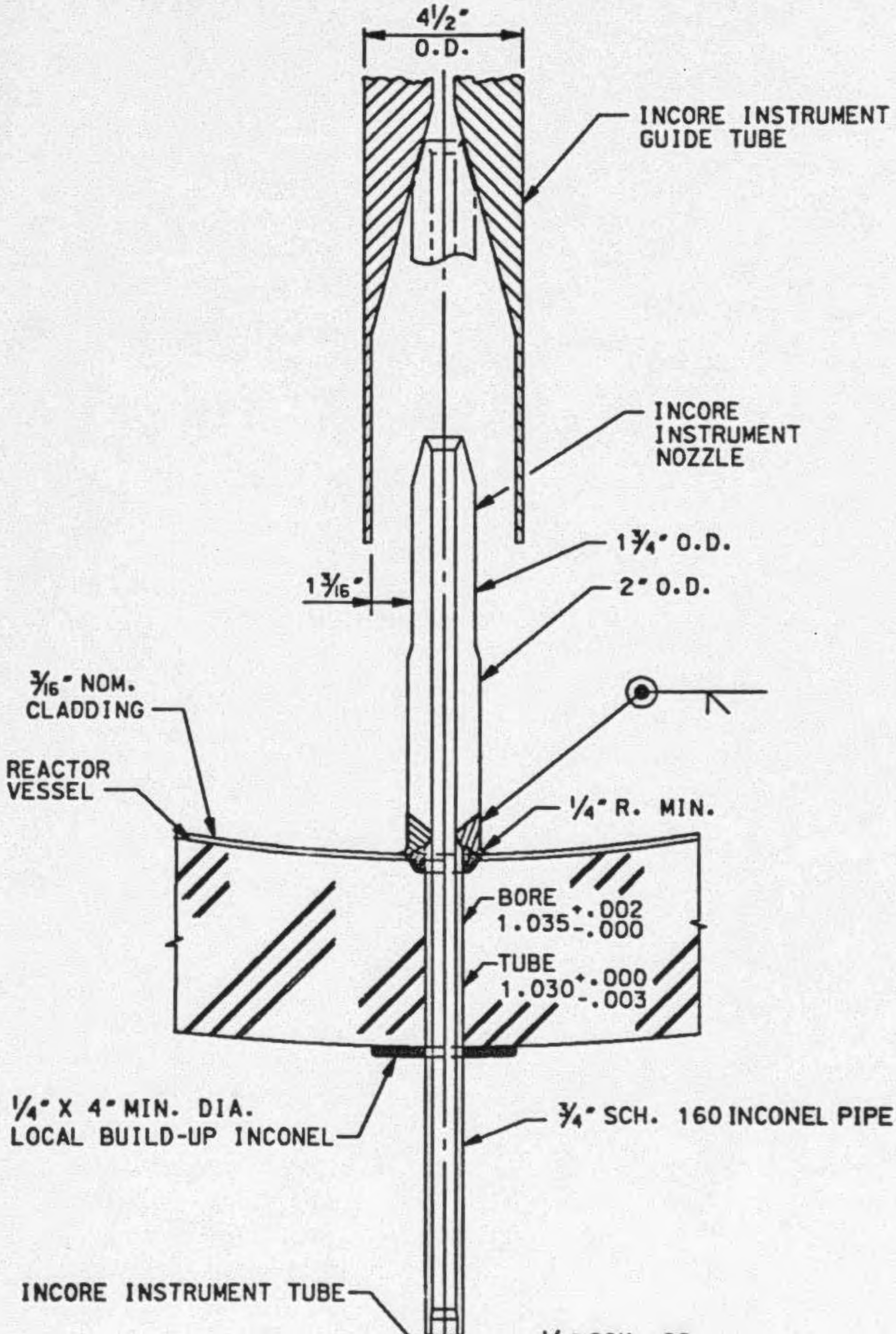
6. "Safety Evaluation Report for Heavy Load Handling Inside Containment", GPU letter 4410-84-L-0193 dated November 1, 1984, F. R. Standerfer to B. J. Snyder.

7. Hazards Analysis: Potential for Boron Dilution of Reactor Coolant System.

8. TMI-2 Technical Planning Bulletin, "Reactor Lower Head Video Inspection", TPB-85-6, Revision 1, March 7, 1985.

9. "Safety Evaluation Report for Plenum Lift and Transfer", GPU letter 4410-85-L-0025, dated January 25, 1985, F. R. Standerfer to B. J. Snyder.

10. Technical Specification Change Request No. 47, GPU letter 4410-85-L-0008, dated February 15, 1985, F. R. Standerfer to B. J. Snyder.



INCORE INSTRUMENT GUIDE TUBE/ NOZZLE INTERFACE

FIGURE 3-1

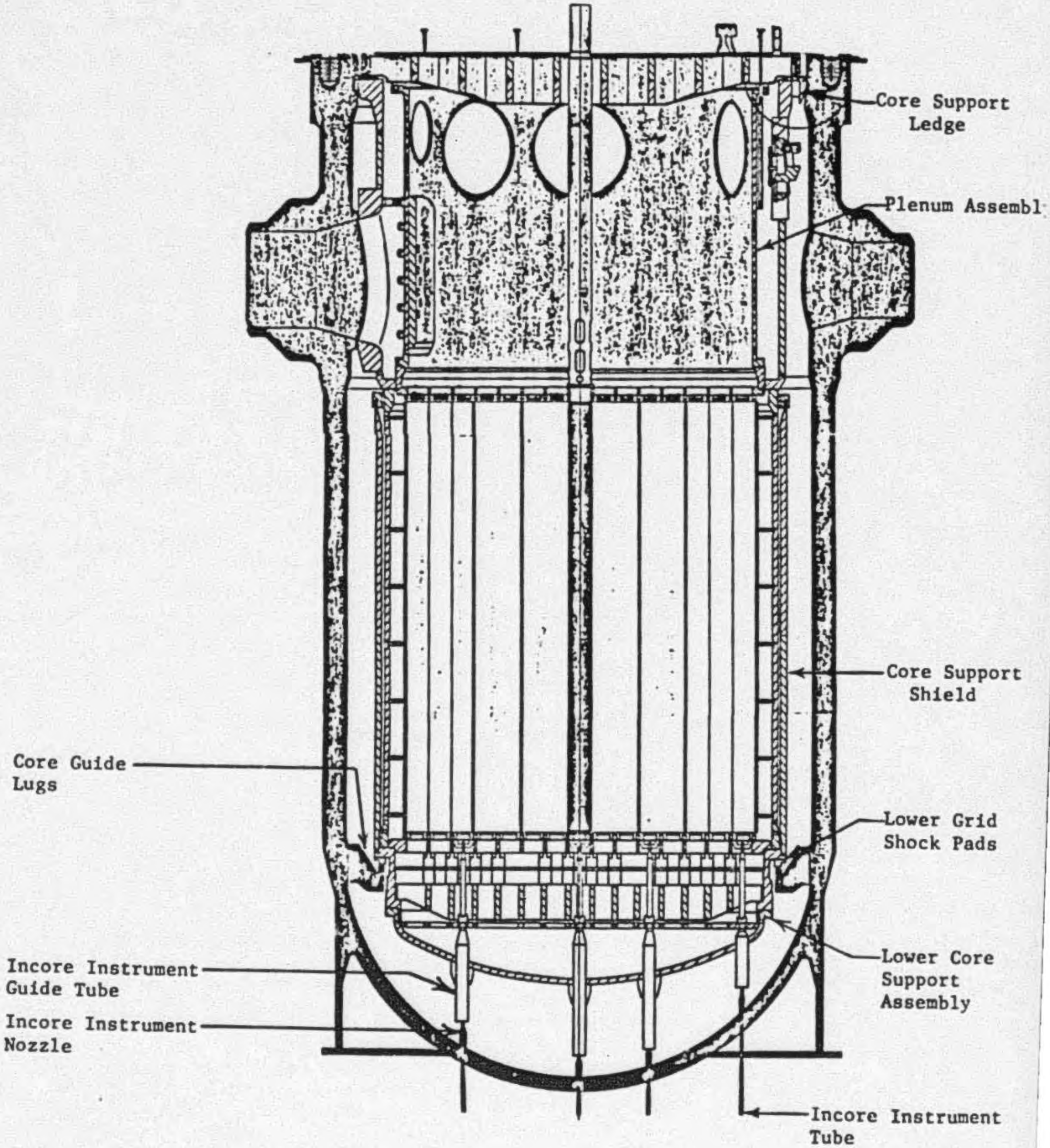

Core Support
Ledge
Plenum Assembl
Core Support
Shield
Core Guide
Lugs
Lower Grid
Shock Pads
Incore Instrument
Guide Tube
Lower Core
Support
Assembly
Incore Instrument
Nozzle
Incore Instrument
Tube

Figure 3-2