GPU Nuclear

SAFETY ANALYSIS

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Page 1

of 19

TITLE

SAFETY EVALUATION REPORT

FOR

LOWER CORE SUPPORT ASSEMBLY DEFUELING

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Rev.	SUMMARY OF CHANGE	Approval	Date
0	Issued for use.	am	2/87
1	Revised to permit the use of unborated coolant water in the plasma arc torch. The torch coolant inventory is limited so that no more than three (3) gallons are able to drain into the Reactor Vessel. Updated the jobhours and person-rem expended for defueling.	am	11/87
2	Revised title of SER to "Safety Evaluation Report for Lower Core Support Assembly Defueling." Revised scope of SER excludes removal of the elliptical flow distributor and gusseted incore guide tubes. Added new Section 4.13, "Reactor Building Basement." Revised Section 4.12, "Heavy Load Drops."	am	1/88

TABLE OF CONTENTS

1.0 PURPOSE AND SCOPE

1.1 Purpose

The purpose of this Safety Evaluation Report (SER) is to demonstrate that the activities associated with defueling the lower core support assembly (LCSA) in the TMI-2 Reactor Vessel and some limited defueling of the TMI-2 Reactor Vessel lower head can be accomplished without jeopardizing the health and safety of the public.

1.2 Scope

This evaluation for LCSA (Figure 1) defueling addresses the following activities:

- o Removal of core debris from within the LCSA, including removal of LCSA structural material when such structural material and core debris are not readily separable.
- o Removal of LCSA structural material (with the exception of the elliptical flow distributor and the gusseted incore guide tubes) to gain access to debris deposits within or below the LCSA.
- o Removal of some core debris from the lower head. Note that lower head defueling by vacuuming was addressed in Reference 1.

LCSA structural material will either be placed in defueling canisters, stored in the Reactor Vessel, stored out of the Reactor Vessel in temporary containers, placed in the Fuel Pool A, Reactor Building basement, or other suitable Reactor Building location pending final disposition.

Equipment expected to be used to support these activities consists of:

- o core bore machine
- o cavitating water jet
- o plasma arc cutting tool
- o Automatic Cutting Equipment System (ACES)
- o robot manipulators
- o equipment/tools as described in Reference 1

As the LCSA defueling operations proceed, the potential exists that activities or equipment described in this report or Reference 1 will need to be modified or new activities and/or tooling developed. Any modifications to existing activities or equipment or the introduction of

new activities or equipment will be reviewed and documented in accordance with TMI-2 administrative procedures to ensure that no potential hazards or safety concerns, not bounded by this SER or Reference 1 are created. If no such hazards or safety concerns are created, LCSA defueling may proceed based on the new or modified activities or equipment without a requirement to revise this SER.

2.0 MAJOR ACTIVITIES AND EQUIPMENT

LCSA defueling will be performed in accordance with detailed approved procedures. Any of the approved activities performed or tools used during initial and/or core region defueling are considered acceptable during LCSA defueling unless specifically precluded. The initial and core region defueling activities and tools are evaluated in Reference 1. Operations to be performed during LCSA defueling include:

- Cutting the LCSA within the Reactor Vessel
- Core debris and structural material removal from the LCSA
- Core debris removal from the Reactor Vessel lower head

2.1 Activities

The current proposed method of dismantling and defueling the LCSA will utilize a core bore machine in conjunction with a plasma arc torch. The dismantling of the LCSA will also provide limited access to the lower head for defueling.

Various combinations of the use of the core bore machine and plasma arc torch may be utilized to obtain the most effective use of each. However, operations will most likely be similar to the following. The first phase of LCSA dismantling and defueling will consist of using the core bore machine to first bore through all 52 incore spiders and then completely bore (i.e., sever from the LCSA) up to 15 outer periphery (non-gusseted) incore guide tubes. Next, the support posts will be bored through to the lower grid forging with 16 outer periphery supports posts being completely bored through the lower grid forging. After these operations are completed, the plasma arc torch will be used to cut the lower grid rib section, lower grid distributor, lower grid forging, incore guide support plate, and any remaining support posts. Each of the sections will be cut one at a time so as to sever the piece from the remaining LCSA structure.

After the respective boring and cutting operations, the severed material will be flushed and removed from the Reactor Vessel. The severed portions of the incore guide tubes, support posts, and possibly other small pieces of the LCSA will be flushed and removed from the Reactor Vessel either in fuel canisters or in special storage containers based on an inspection of the material for visible fuel. The sections of the

lower grid rib section, lower grid distributor, lower grid forging, and incore guide support plate will be lifted, flushed, inspected to ensure no visible fuel is present, removed from the Reactor Vessel and stored in the Reactor Building basement or other suitable Reactor Building location pending final disposition.

The exact sequence of operations shall not be limited to that described above. Changes in operation sequence will not necessitate a revision to this SER unless safety concerns created by the change are not bounded by this SER or Reference 1.

2.2 Equipment

Descriptions of tools required for LCSA defueling are provided below.

The Core Bore Machine

A detailed description of the core bore machine was previously provided in References 2 and 15. The core bore machine is a self-contained machine consisting of a hydrostatically driven spindle and hydraulically actuated feed cylinders. The spindle is equipped with a hydraulically actuated drill chuck in order to apply controlled rotational torque and speed to the drill string. The cylinders are used to apply the downward force on the drill bit and to move the drill string into and out of the drilled hole. Initial operations with the core bore machine will utilize various new tool holders, core liners and cutters specifically designed to cut the incore guide tubes, and support posts from the LCSA.

Plasma Arc Torch

The plasma arc torch is provided to cut electrically conductive materials, such as stainless steel structures, which inhibit access to fuel to be removed. The torch will be operated via the remote manipulator or other positioner to allow remote operation of the torch. The torch is a direct current arc, tungsten electrode, metal burning device. An initial pilot arc will ionize the primary gas, nitrogen, to form a plasma jet. A secondary gas, also nitrogen, is used to aid in flushing away the molten metal from the cut and to provide insulation for the torch head. The total gas flow is approximately 20 scfm. Testing of the plasma arc torch has shown that Reactor Coolant System (RCS) grade or B-10 enriched borated water cannot be used as the torch coolant due to the high electrical conductivities of these fluids. Further testing has determined that the use of demineralized (i.e., unborated) water results in acceptable torch operation. Consequently, unborated water will be used as the cooling fluid for the torch. The torch coolant inventory is limited so that no more than three (3) gallons of unborated water are able to drain into the Reactor Vessel. Criticality concerns associated with the use of unborated coolant water are addressed in Section 4.2 of this SER.

Cavitating/Pulsating Water Jet System/Flushing System

The cavitating/pulsating water jet system is provided to erode fuel debris from metal surfaces within the Reactor Vessel and to break up large debris pieces to facilitate removal. The system will flush tightly adherent debris from vessel structures and will break up the fuel debris into particles amenable to vacuuming. The system consists of high pressure discharge pumps (approximately 6-15 gpm at 10,000-20,000 psi), cavitating jet nozzles and lances, and connecting hoses and piping. The pumps will be mounted on the 347'-6" elevation, be powered by electric motors and take suction from the Defueling Water Cleanup System (DWCS) suction. Any potential siphoning of the Reactor Vessel inventory as a result of a line break upstream of the pump is limited by the safety systems inherent in the Defueling Water Cleanup System (DWCS) (Reference 3). Piping downstream of the pump is precluded from siphoning because it is fixed above the RCS water level. The cavitating/pulsating water jet system will be operated using the remote manipulator or other positioner to allow remote manipulation of the device.

Automated Cutting Equipment System

The Automated Cutting Equipment System (ACES) will position the plasma arc torch to cut LCSA structural elements. The equipment that will operate in the vessel is: a support frame that provides x-y positioning, a manipulator arm that provides vertical travel, rotation, angular positioning, with the ability to grip, release and position the plasma torches. The in-vessel components are powered by a modified train of three commercially available plasma power supplies and one ACES power supply, and operated by a control system. The computerized control system is capable of controlling all five axes of the in-vessel equipment and can locate the torch nozzle and move it over a pre-determined path at controlled rates. The very important cutting parameter, torch to work distance, is controlled continuously and automatically by a servo motor and feed back loop taking its signal from the torch arc voltage. All of the torch operations are pre-programmed after verification of the program modeled to the in-vessel LCSA. The controller is located in a Command Center outside of the containment building and is supported by a computer-assisted-design model of the LCSA. The operators are assisted with both video monitor and printer output. In addition to ACES, a manual positioner is also available for torch positioning.

Robotic Manipulator

Two (2) hydraulicly operated manipulator arms will be mounted on the Manual Tool Positioner (MTP) or other suitable mast. One of the manipulators (Grabber) can be used to stabilize the MTP while the other manipulator (Work) is used to help remove debris and structural material after it has been cut. The manipulators will have a separate borated

hydraulic power supply and will be normally operated from outside the Reactor Building although the ability to operate from within the Reactor Building exists and may be utilized.

Mechanical Tools

Mechanical Tools will be used to cut structural material (abrasive saw) and prepare structural material for the plasma arc torch (grinder/milling tools). Some tools will be powered by a borated hydraulic power supply.

Other tools and equipment are as described in Reference 1.

3.0 COMPONENTS AND SYSTEM AFFECTED

Other components or systems in addition to those described in Reference 1 may be required to conduct the LCSA defueling activities. Where this is the case, the use of the component or system will be evaluated to ensure that its use is bounded by the evaluations of this SER or Reference 1.

4.0 SAFETY CONCERNS

4.1 General

An evaluation of the activities associated with LCSA defueling identified the following safety concerns:

- RCS Criticality Control
- Boron Dilution
- Hydrogen Evolution
- Pyrophoricity
- Submerged Combustion
- Fire Protection
- Decay Heat Removal
- Instrument Interference
- Release of Radioactivity
- RV Integrity
- Heavy Load Drops
- Reactor Building Basement

Each of these issues are discussed below.

4.2 RCS Criticality Control

The evaluations provided by References 1, 4, and 5 generally bound this concern during LCSA defueling. The torch cooling system has, by design, a maximum unborated coolant inventory of less than four (4) gallons. However, the maximum amount of unborated water that could drain into the Reactor Vessel from the coolant system when it is in its operating position is no more than three (3) gallons. As this quantity of

unborated water exceeds the previous limit of two (2) gallons established in Reference 5, a criticality analysis was performed to demonstrate that the use of unborated coolant for the plasma arc torch would not pose a criticality safety concern. Reference 6 provides the basis, assumptions, and bounding fuel models used in the plasma arc torch criticality analysis. Based on the results of the analysis, it is concluded that the plasma arc torch, with a maximum drainable coolant system inventory of three (3) gallons of unborated water, can be used to dismantle the LCSA without developing a criticality safety concern within the Reactor Vessel.

The above conclusion is based on the operational limitations listed in Reference 6.

4.3 Boron Dilution

Boron dilution concerns during LCSA defueling are bounded by the evaluations provided by References 1 and 7. To preclude the possibility of a hydraulic fluid leak leading to a possible critical configuration of fuel and moderator, all hydraulic fluid used with LCSA defueling tools with the exception of the core bore machine will be borated to at least 4350 ppm natural boric acid. The hydraulic fluid in the core bore machine does not need to be borated as there is no potential for it to mix with the fuel (Reference 2).

4.4 Hydrogen Evolution

Small quantities of hydrogen gas generation (less than 1/10 SCFM) will be a by-product of the plasma arc cutting tool operation underwater. This hydrogen will be diluted by the off-gas treatment system, as required, and thus, a combustible concentration will not occur within the Reactor Building. Other hydrogen related safety issues are bounded by the evaluations provided in Reference 1.

4.5 Pyrophoricity

Pyrophoricity concerns during LCSA defueling are bounded by evaluations provided in References 1 and 8.

4.6 Submerged Combustion

The use of underwater burning devices (e.g., plasma arc torch) creates a heat source not previously considered. This additional heat source is not expected to create a combustion concern since the plasma arc torch will be operated underwater. Additionally, testing of thermic torch and plasma arc burning devices on alumina filled zirconium tubes underwater did not produce any sustained ignition (Reference 9 and 10). It is considered reasonable not to postulate a combustion reaction of exposed fuel debris due to operation of the plasma arc torch.

4.7 Fire Protection

The evaluation provided by Reference 1 bounds this concern during LCSA defueling.

4.8 Decay Heat Removal

Decay heat removal concerns during LCSA defueling are generally bounded by the evaluation provided in Reference 1. The maximum power requirements for the plasma arc torch are 1000 amps at 200 volts DC. Operation of the torch underwater will provide a significant heat source; however, continuous operation is not probable due to the need to reposition the torch. Even if the torch were to operate continuously for one hour, it would raise the RCS temperature only approximately two (2) degrees. The RCS temperature will be monitored to preclude an uncontrolled water temperature increase.

4.9 Instrument Interference

Issues regarding instrument interference caused by the use of the plasma arc torch are bounded by the evaluation provided in Reference 10.

4.10 Release of Radioactivity

The central zone of the plasma arc reaches temperatures of 20,000°F to 50,000°F and is completely ionized. However, this high energy is quickly dissipated and primarily heats the conductive metal. It is expected that fuel on the metal surfaces will also be heated to the liquid or vapor state. Most fuel so heated will immediately oxidize, transfer its heat to the surrounding water, resolidify and sink. Soluble isotopes trapped in the fuel matrix may become dissolved in the water. This possible increase in the concentration of radioactivity is not expected to be prohibitive or exceed that observed in the core drilling program. Safety concerns associated with the release of radioactivity from the Reactor Vessel to the environment are bounded by the evaluations in Reference 1.

4.11 Reactor Vessel Integrity

Damage to the Reactor Vessel due to the operation of burning devices inside the vessel is precluded. The operation of such devices is physically limited to inside the confines of the core support structure and the elliptical flow distributor where the torch is more than one-foot away from the Reactor Vessel wall. Cutting operations will begin on the top of the LCSA and will sequentially cut through the lower grid rib assembly, lower grid flow distributor, lower grid forging, and incore guide support plate. Therefore, the arc or flame of such burning devices, operating underwater, will always be operated at least a foot from the Reactor Vessel wall. Propagation of an arc through one foot of water is not possible, thus, damage to the Reactor Vessel wall due to the operation of burning devices is precluded.

The only other major concern associated with Reactor Vessel integrity is in regards to the integrity of the incore nozzles. Previous GPU Nuclear and NRC correspondence (Reference 11 and 12) established two (2) possible incore nozzle configurations as a result of the 1979 accident. In the worst case, the damage to the Reactor Vessel lower head would consist of an incore nozzle melted to the inside diameter of the Reactor Vessel lower head with a nozzle to vessel weld thickness of only 0.030". The significance of this configuration is that if the weld experienced significant damage, the incore nozzle above the weld would have melted. The other possible configuration is that the incore nozzle was undamaged.

It is highly unlikely that the incore nozzles were degraded during the accident. This is especially true of those incore nozzles located under the peripheral non-gusseted incore guide tubes. Based on the qualitative assessment provided by References 15 and 16, it was concluded that the incore nozzles in this area are unlikely to be degraded. However, due to the inability to visually inspect all the incore nozzles, it must be assumed that on a worst case basis, the incore nozzles under at least some of the gusseted incore guide tubes experienced some degree of damage. Therefore, care must be exercised when defueling the LCSA.

The use of tools that could potentially impart excessive loads to the incore instrument tube nozzles or damage the Reactor Vessel wall will be limited to use within the confines of the core support structure and the elliptical flow distributor until most of the fuel within the lower LCSA has been removed after which procedural limitations will be applied. Mechanical cutting devices, such as the abrasive saw, grinding wheel, and impact hammer are not of sufficient size or power to damage the Reactor Vessel wall and, therefore, do not create a safety issue.

During the removal of fuel debris from the lower head, care will be exercised to prevent excessive loads on exposed incore nozzles. If, during the process of removal of fuel in the vicinity of an incore nozzle, observations indicate that the nozzle has suffered damage due to excessive temperatures, work will be halted and the situation evaluated to ensure that activities can continue within the bounds of this SER.

Other Reactor Vessel integrity safety concerns (e.g., assessment of potential damage to incore nozzles from pulling on incore instrument strings) are bounded by the evaluations provided in Reference 12.

4.12 Heavy Load Drops

During LCSA defueling, the Reactor Vessel lower head and the incore nozzles under the non-gusseted incore guide tubes will be subject to potential direct load drops. Calculations have demonstrated that a potential load drop through a hole in the elliptical flow distributor onto an undamaged incore nozzle would not result in a nozzle failure.

In order to preclude the potential for a load drop on a potentially damaged incore nozzle, the elliptical flow distributor plate and the gusseted incore guide tubes will be left in place for this phase of LCSA disassembly and defueling.

As mentioned previously, the sections of the LCSA will be cut into pieces using the plasma arc torch. After the LCSA pieces have been cut from the main LCSA structure, they will be rigged and lifted out of the Reactor Vessel. The heaviest pieces to be removed are the two (2) halves of the lower grid forging. These pieces will weigh approximately 9,000 pounds. Should the forging drop from an elevation just above the guard rails of the shielded work platform, it would fall slightly less than 50 feet to impact the incore guide support plate. The shear stress developed within the smallest ligament of the incore guide support plate due to impact would exceed the ultimate stress of the plate material. However, the energy of the fall would likely be absorbed in the support plate leaving little excess energy to cause a similar failure of the elliptical flow distributor. Consequently, the integrity of the Reactor Vessel would be expected to be maintained. As an alternate to this removal procedure, the grid forging may be cut into smaller pieces to enhance removal from the vessel and facilitate storage. Should this option be used, the weight of the assumed falling plate would be less than stated above. Further, it is assumed that since other pieces of the LCSA will weigh less than the lower grid forging, the dropping of such pieces onto the remaining LCSA structures, which are identical or stronger than the incore guide support plate; will not result in structural failure of the LCSA.

When the pieces of the grid forging are removed, they will have to be lifted over the upper tubes of the remaining incore guide tubes. The remaining incore guides tubes, however, will be bolted to the incore guide support plate and welded to the elliptical flow distributor. Since the shear strength of the gusset welds, elliptical flow distributor welds, and the bolting at the support plate exceed the compressive strength of the 1 15/16" diameter incore guide tube above the support plate, shearing the incore guide tube from the elliptical flow distributor is not likely. Further, based on the small target provided by an incore guide tube stub above the support plate an impact from above the incore guide tube stub would likely cause it to bend at or near the nut holding the guide tube into the incore guide support plate. Therefore, it may be concluded that the incore guide tube cannot impact the Reactor Vessel lower head or a degraded incore nozzle as it will remain welded to the elliptical flow distributor.

Removing all the plates above the elliptical flow distributor will open many 6" diameter holes through which direct access to the Reactor Vessel lower head is obtained. However, based on previous analysis (Reference 15 and 17), it was shown that potential load drops of objects less than 6" in diameter, including dropped incore guide tubes, would not impart excessive stresses to intact incore nozzles. Further, with the

elliptical flow distributor in place objects less than 6" diameter, should they fall through a hole in the elliptical flow distributor, would not impact a potentially degraded incore nozzle since these nozzles are not in line with the open holes. The potentially degraded incore nozzles are also shielded from dropped objects by the gusseted incore guide tubes which still remain in the elliptical flow distributor. Consequently, it is concluded that during the LCSA disassembly and defueling process dropped objects cannot strike potentially degraded incore nozzle welds and as a consequence the potential for Reactor Vessel leakage due to dropped loads is remote.

The potential for load drop accidents into the Reactor Vessel is also minimized by careful control of load handling activities and the use of load handling equipment which has been conservatively designed and tested. Load handling activities are performed in accordance with approved procedures for such activities including 4000-PLN-3891.02, "TMI-2 Lifting and Handling Program." Each specific load handling activity is controlled by a Unit Work Instruction or procedure. Load handling activities will be performed by personnel who have been trained and qualified for these activities.

4.13 Reactor Building Basement

The above evaluation concludes that dropped objects will not cause Reactor Vessel leakage, however, for completeness, the potential effect of Reactor Vessel leakage was also considered. The case considered assumed the failure of one (1) nozzle resulting in a 125 GPM leak from the Reactor Vessel. As previously documented (most recently in Reference 16), such an event would be promptly detected and capabilities exist to maintain RCS level. However, due to the recent defueling tasks, a large accumulation of small fuel bearing particles have been deposited into the Reactor Vessel lower head. Consequently, it is assumed that a portion of the fuel debris in the lower head would migrate into the basement cavity below the Reactor Vessel should a Reactor Vessel leak occur.

An analysis was performed to evaluate the criticality safety concerns associated with the relocated fuel within the reactor cavity. The results of this analysis, which was performed using optimum geometry and moderation, are provided in Appendix A. The analysis concludes that the maximum allowable fuel mass within the cavity assuming a 2950 ppm boron concentration is approximately 40,000 lbs. It is considered incredible that this much fuel could be relocated to the cavity via the failure of a degraded incore nozzle. Consequently, it is concluded that maintaining the boron concentration of any water within the reactor cavity above 2950 ppm will eliminate a criticality safety concern as a result of fuel relocating to the reactor cavity.

To ensure that the reactor cavity water boron concentration will always remain above 2950 ppm a sample pump will be installed in the cavity and the following general approach will be used:

- o Add enough borated water to the reactor cavity to increase the boron concentration within the cavity region to at least 3500 ppm.

- o Sample the water in the cavity region weekly or whenever Reactor Building water level increases more than three (3) inches since the previous sample. This increase is assumed to be a result of other activities within the Reactor Building (e.g., decontamination).

- o After sampling, if the boron concentration is below 3500 ppm, add enough borated water to increase boron concentration to at least 3500 ppm.

- o Repeat the above steps until the Reactor Building basement water level reaches an elevation of 283.40'. At this point, the Reactor Building water level must be reduced and the cavity region reborated to at least 3500 ppm.

If boron concentration in the cavity beneath the Reactor Vessel falls below 2950 ppm, Core Alterations will be suspended until the concentration is restored. Because this evaluation concludes that Reactor Vessel leakage will not occur, the controls discussed above are not required. However, for conservatism, prior to removal of the lower grid forging, these controls will be implemented.

5.0 RADIOLOGICAL CONSIDERATIONS

Based on a comparison of activities associated with Reference 1 to those associated with LCSA defueling, it is concluded that the radiological considerations associated with LCSA defueling are bounded by Section 5 of Reference 1. However special precautions will have to be taken to prevent exposure of operating personnel during transport of radioactive and contaminated pieces of the LCSA from the Reactor Vessel to the storage location. The present plan discussed in Section 2.1 is to flush all LCSA pieces prior to removal from the Reactor Vessel in order to remove the visible fuel debris and reduce radioactive contamination on each piece.

It is expected that the pieces that were nearest the core will be the most radioactive due to Co-60 activation. Thus, the lower grid rib assembly will be the most highly radioactive piece removed from the Reactor Vessel. The estimated radiation level of an 8' x 8' piece of the lower grid rib assembly is 280 rem/hr at one (1) foot. At a distance of 30 feet, the radiation level is 2.1 rem/hr. This plate will have to be rigged, moved, and unrigged, if necessary, remotely. The other sections of the LCSA will represent less of a radiation hazard. Personnel exposure control during the handling of these pieces will be in accordance with approved ALARA practices.

An update of the jobhours and person-rem expended to date for all defueling activities is provided in Table 5.1. The overall estimated occupational exposure to complete Reactor Vessel defueling remains at approximately 1400 person-rem.

TABLE 5.1

Jobhours and Person-rem Expended Through October 1987

Activity	Jobhours	Person-rem
Preparations, installations	3,930	100
Operations	34,899	343
Maintenance/Support	20,570	311
Decontamination and Removal*	0	0
TOTALS	59,399	754

*No activity associated with final decontamination and removal of defueling equipment has been performed as of January 1, 1988, thus no jobhours and person-rem are given. Note, decontamination maintenance in the Reactor Building is not considered part of this activity.

6.0 IMPACT ON PLANT ACTIVITIES

The major potential impact of LCSA defueling on plant activities is the effect of fuel movement in Unit 2 on operations in Unit 1. Based on the evaluation provided in Reference 1 and the similarity of the activities considered in Reference 1 to those activities within the scope of this SER, it is concluded that the LCSA defueling operations in Unit 2 will not affect personnel in Unit 1.

7.0 10 CFR 50.59 EVALUATION

10 CFR 50, 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.

10 CFR 50, Paragraph 50.59, states a proposed change involves an unreviewed safety question if:

a. The probability of occurrence or the consequence 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.

Although there are notable differences between the proposed defueling activities for TMI-2 and routine activities described in the FSAR, the consequences of postulated accidents are not different and as demonstrated in Reference 1, are sufficiently similar to be compared. Reference 1 compared two (2) potential events during defueling, a canister drop accident and a Krypton 85 release, to two (2) events described in the FSAR, a fuel handling accident and a waste gas decay tank failure. The comparison demonstrated that on a worst case basis, the consequences of the FSAR events bound the consequences of any defueling-related event.

A variety of postulated events were analyzed in this SER for LCSA defueling. The analysis of these events provided in Section 4 results in the conclusion that the postulated events are also bounded by previous evaluations and/or do not result in an unanalyzed condition.

To determine if LCSA defueling activities involve an unreviewed safety question, the following questions must be evaluated.

Has the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report been increased?

A variety of events were analyzed in Reference 1. It was demonstrated that these events were bounded by comparable events analyzed in the FSAR. It was shown that the potential consequences from these events were substantially less than the potential consequences of comparable events analyzed in the FSAR. Section 4 of this SER demonstrates that the consequences of potential events during LCSA defueling are bounded by previous evaluations.

During this phase of LCSA disassembly and defueling, the degraded incore nozzles will be protected from a load drop, therefore, the potential for a leak due to a load drop is not increased. Additionally, because a Reactor Vessel leak is not likely, the potential for fuel fines from the Reactor Vessel to migrate to the cavity beneath the Reactor Vessel in the Reactor Building basement due to an incore nozzle failure is remote. Section 4.13 of this SER demonstrates that a basement criticality event due to the presence of this fuel is prevented because of the boron concentration that will be present in the cavity.

By considering postulated events and reviewing various safety mechanisms, i.e., fire protection and decay heat removal, it has been demonstrated that LCSA defueling activities will not adversely effect equipment classified as important to safety (ITS). Consequently, it is concluded that the probability of a malfunction of ITS equipment or the consequences of a malfunction of ITS equipment has not been increased.

Therefore, it is concluded that the proposed activities associated with LCSA defueling do not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report.

<u>Has the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report been created?</u>

The variety of postulated events analyzed in Reference 1 considered a spectrum of event types which potentially could occur as a result of the defueling process. A comparison of those events with comparable events in the FSAR demonstrated that the event types postulated for the defueling process are similar and bounded by the FSAR. In addition, no new event type was identified which was different than those previously analyzed in the FSAR. Section 4 of this SER evaluates events postulated for LCSA defueling. These type of events have been previously evaluated and, therefore, do not represent a different type of accident or malfunction.

<u>Has the margin of safety, as defined in the basis for any technical specification been reduced?</u>

Technical Specification safety margins at TMI-2 are concerned with criticality control and prevention of further core damage due to overheating. Technical Specification safety margins will be maintained throughout the LCSA defueling process. Subcriticality is ensured by establishing the RCS boron concentration at greater than 4350 ppm or equivalent and ensuring that this concentration is maintained by monitoring the boron concentration and inventory levels and by isolating potential deboration pathways. Systems will remain in place to add borated cooling water to the core in the event of an unisolable leak from the Reactor Vessel to prevent overheating and potential criticality. Boron will be added to the cavity beneath the Reactor Vessel, prior to removal of the lower grid forging, to ensure that in the event of a leak, a criticality event external to the vessel is not credible. The introduction of unborated water from the torch cooling system will not create the potential for a criticality.

No Technical Specification changes are required to conduct the activities bounded by this SER.

In conclusion, the LCSA defueling activities do not:

- o Increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report, or

- o Create the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report, or

- o reduce the margin of safety as defined in the basis for any technical specification.

Therefore, the LCSA defueling activities do not constitute an unreviewed safety question.

8.0 ENVIRONMENTAL ASSESSMENT

Based on Section 8.0 of Reference 1 and noting the similarities between the activities considered in Reference 1 to those activities within the scope of this SER, it can be concluded that the proposed LCSA defueling activities can be performed with no significant environmental impact.

9.0 CONCLUSIONS

Activities associated with LCSA defueling have been described and evaluated. Based on the results of this analysis, it can be concluded that the consequences of postulated accidents with respect to potential core disturbances will not compromise plant safety. The evaluations have also shown that the tasks and tooling employed follow the continued commitment to maintain radiation exposure levels ALARA. Therefore, it is concluded that LCSA defueling activities can be performed without presenting undue risk to the health and safety of the public.

10.0 REFERENCES

1. Safety Evaluation Report for Defueling the TMI-2 Reactor Vessel, Revision 10, 15737-G07-108, May 1986.

2. Safety Evaluation Report for Core Stratification Sample Acquisition, Revision 4, 15737-2-G07-109, July 3, 1986.

3. Technical Evaluation Report for Defueling Water Cleanup System, Revision 8, 15737-2-G03-106, December 1985.

4. Criticality Report for the Reactor Coolant System, Revision 0, 15737-2-N09-001, October 1984.

5. Report on Limits of Foreign Materials Allowed in the TMI-2 Reactor Coolant System During Defueling Activities, Revision 1, 15737-2-N09-002, September 1985.

6. Criticality Safety Assessment for Using the Plasma Arc Torch to Cut the LCSA, 15737-2-N09-004, November 1987.

7. Hazards Analysis: Potential for Boron Dilution of Reactor Coolant System, Revision 2.

8. GPU TPO/TMI-127, Revision 0, "Technical Plan for Pyrophoricity," December 1984.

9. EG&G Plasma Arc Test Report, LCSA-4, April 30, 1986.

10. GPU Nuclear letter to W.D. Travers, USNRC, 4410-86-L-0143 dated August 27, 1986, Subject: Use of Plasma Arc Torch.

11. GPU Nuclear letter 4410-86-L-0162, dated September 19, 1986, "Core Bore Operations," to W.D. Travers from F.R. Standerfer.

12. NRC letter NRC/TMI-86-01, dated October 16, 1986, "Core Bore Operations," to F.R. Standerfer from W.D. Travers.

13. GPU Nuclear letter 4410-87-L-0160, dated December 3, 1987, "Use of Core Bore Machine for Dismantling the Lower Core Support Assembly," to DCD from F.R. Standerfer.

14. Reactor Building Sump Criticality Safety Evaluation Report, Revision 2, 4550-3254-85-02, January 1986.

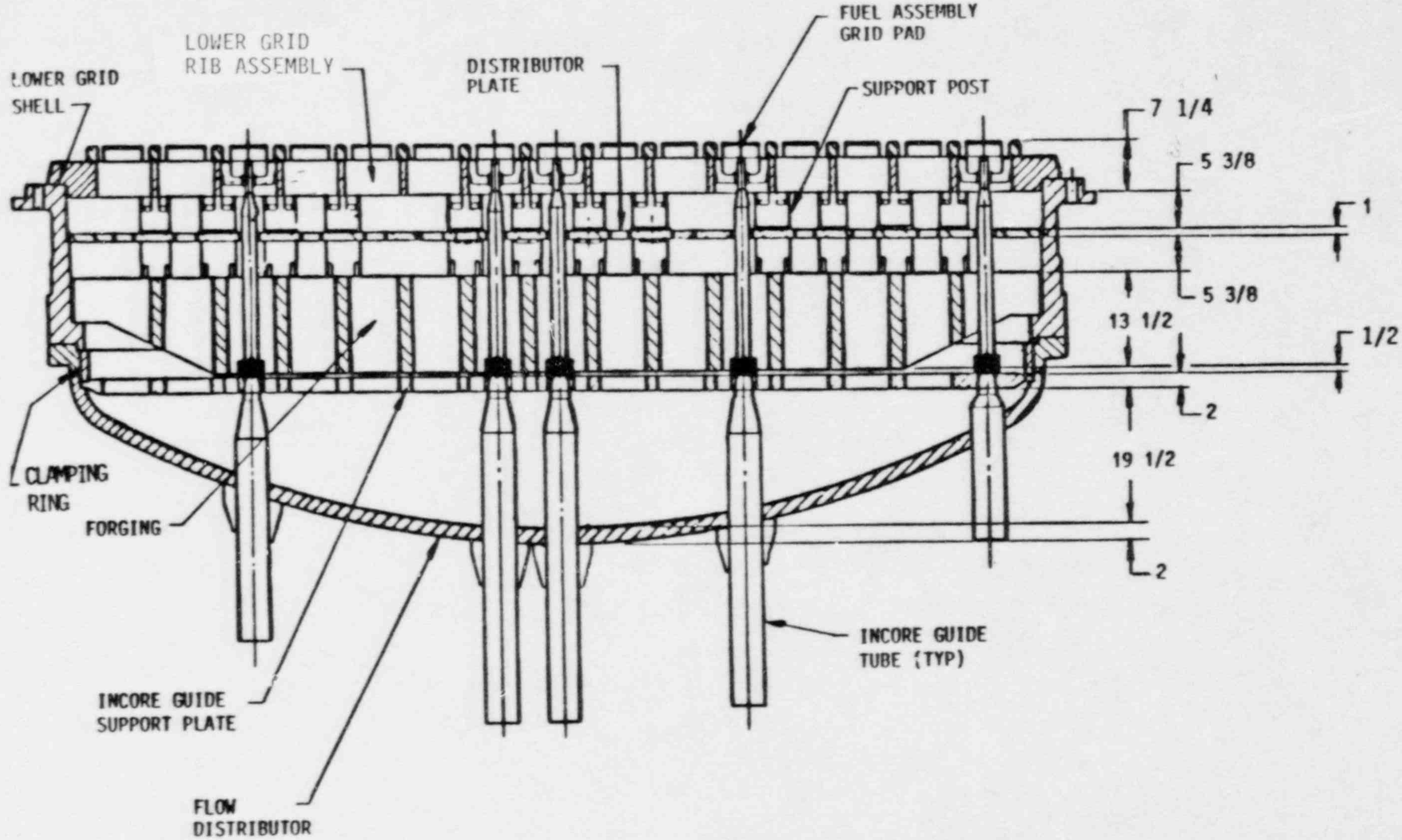
15. GPU Nuclear letter 4410-87-L-0189 dated December 28, 1987, "Use of Core Bore Machine for Dismantling the Lower Core Support Assembly."

16. NRC Letter NRC/TMI 88-003 dated January 8, 1988, to F. R. Standerfer from W. D. Travers.

17. Safety Evaluation Report for Core Support Assembly and Lower Head Defueling, Revision 0, 4710-3221-86-011, February 1987.

FIGURE 1
4410-88-L-0005

Cross Section of Lower Grid, Flow Distributor and Guide Tube Assembly



1.0 INTRODUCTION

During the cutting and/or removal operations associated with Lower Core Support Assembly (LCSA) or lower head defueling it can be postulated that loads will be dropped onto the incore guide tube nozzles. These loads could be incore guide tubes or portions of the LCSA which have been cut with the core bore machine or the plasma arc torch. These dropped loads are postulated to damage the incore nozzle-to-reactor vessel weld, driving the nozzle from the vessel. This arrangement may create an opening for fuel fines and borated water to flow out of the vessel and into the reactor building basement. The intent of this report is to evaluate the criticality safety consequences associated with fuel being located on the basement floor inside the reactor cavity.

2.0 CRITICALITY SAFETY MODEL

The following paragraphs describe the modelling used in the criticality safety analyses of this report. In general, the approach used for these analyses was to determine the maximum amount of fuel that could safely be located on the reactor building basement floor assuming various boron levels in the basement water. The boron concentrations assumed in the analysis were 1500, 2000, 2500 and 3000 ppm.

2.1 FUEL COMPOSITION

As with previous criticality safety analyses performed for TMI-2 (References 1,2,3), the fuel was represented as a homogeneous medium for which the neutronic data corresponded to a dodecahedral lattice structure of spherically shaped fuel pellets. The fuel particle size was assumed to be the equivalent of standard pellets. This assumption was based in part on the limited size of openings assumed to be in the vessel's lower head. The size of an opening in the vessel is expected to range from a small crack up to the size of an incore nozzle. Most of the fuel debris that exits the vessel should be fines and small particles. Substantial quantities of large particles (i.e., standard pellets size and greater) are not expected to be flushed out of the vessel via the small opening created by the loss of an incore nozzle. Additionally, analyses presented in Reference 3 show that the impurities present in the melted fuel particles (e.g., fuel larger than standard pellets) can have a significant negative reactivity effect.

The enrichment of the fuel used in the current evaluation corresponded to a homogeneous mixture of all three fuel batches. The initial core loadings were used to determine the relative percentages of each batch. The effects of burnup for all three fuel batches was also considered. The technique used to determine the burnup effects is presented in Reference 3. The resultant U-235 enrichment was 2.24%. This enrichment is considered conservative, in that prior to LCSA defueling most of the highest enriched batch 3 fuel (~75%) was removed from the vessel.

2.2 GEOMETRY

It is postulated that any fuel that drains from the reactor vessel will accumulate on the basement floor in an inverted cone or hemispherical shape. For criticality safety purposes, these arrangements can be conservatively approximated as a cylinder. Preliminary efforts have shown that the most reactive cylinder size occurs when the diameter is approximately equal to the height of the cylinder (i.e., a square cylinder). These preliminary efforts have also indicated that the neutron leakage associated with a square cylinder is essentially equal to the neutron leakage from a sphere of the same volume. Consequently since a spherical model, because of the radial symmetry, allows the use of the one-dimensional computer code XSDRNPM, spheres rather than cylinders were used for the finite geometry analyses reported.

Based on the above paragraph, a simple spherical model was developed for this report. In this model the fuel that drains to the basement was assumed to be optimally mixed with the borated basement water. This fuel-water mixture is modelled as a sphere and then surrounded with an infinite borated water reflector. The boron level in the reflector is assumed equal to that in the fuel-water mixture. Four basement boration levels were evaluated - 1500, 2000, 2500, and 3000 ppm. These boron concentrations may be conservatively considered to be the initial basement boron concentration or may represent the increased basement boron level due to the addition of the borated RCS water.

2.3 CONSERVATISMS

Conservatisms inherent in the above described model include the following:

- The fuel is assumed to be a mixture of all three fuel batches, with relative percentages of each batch being based on initial core loadings.

- o The fuel is conservatively assumed to accumulate in the shape of a square cylinder. In reality the fuel debris is expected to be widely distributed within the cavity.
- o No credit is taken for the presence of large amounts of structural material or solid poison in the debris.
- o An optimized fuel tc moderator ratio is used.
- o Standard fuel pellets are used, whereas actual fuel particles are likely to be fines and small particles. The small particles have been shown to have a smaller reactivity worth.
- o No credit is taken for the localized effect of highly borated (4950 ppm) RCS water draining from the vessel along with any fuel. Only a single, basement average boron concentration is considered in the analyses.
- o Enough water is assumed to be present in the basement such that the fuel is completely immersed. If this were not the case, a reduction in the neutron multiplication would occur as some of the fuel would be without adequate moderation.

3.0 RESULTS

The first set of results for these analyses is given in Table 1. This table provides the optimum volume fraction and k_∞ data for each of the boron concentrations of interest (Reference 4). These volume fractions were used in the finite geometry analyses described above.

The results of the finite geometry analyses for each of the boron concentrations are shown in Table 2. This table provides the amount of fuel debris that can drain from the vessel onto the basement floor and not pose a criticality safety concern for various assumed boron concentrations in the basement water. The basis for determining that a particular amount of fuel should be considered safe was that the calculated neutron multiplication (k-eff), including a 2.5% Δk analytical uncertainty bias, would not exceed 0.99. These results are also provided in graphical form in Figure 1.

4.0 REFERENCES

1. Report on Limits of Foreign Materials Allowed in the TMI-2 Reactor Coolant System During Defueling Activities, Rev 1.,15737-2-N09-002, September 1985.

2. Criticality Report for the Reactor Coolant System, Rev. 0, 15737-2-N09-002, October 1984.

3. Criticality Safety Assessment for Using the Plasma Arc Torch to Cut the Lower Core Support Assembly, Rev 1., 15737-2-N09-004, November 1987.

4. Letter, C.V. Parks (ORNL) to D.S. Williams (GPUN), November 30, 1987.

Table 1

Boron Concentration (ppm)	Optimum Volume Fraction	k_∞
1500	0.49	1.0992
2000	0.52	1.0608
2500	0.55	1.0291
3000	0.58	1.0021

All results are provided via Reference 4.

Table 2

Boron Concentration (ppm)	Maximum Fuel Mass (lbm)
1500	4179
2000	7768
2500	17007
3000	44931

All results are provided via Reference 4.

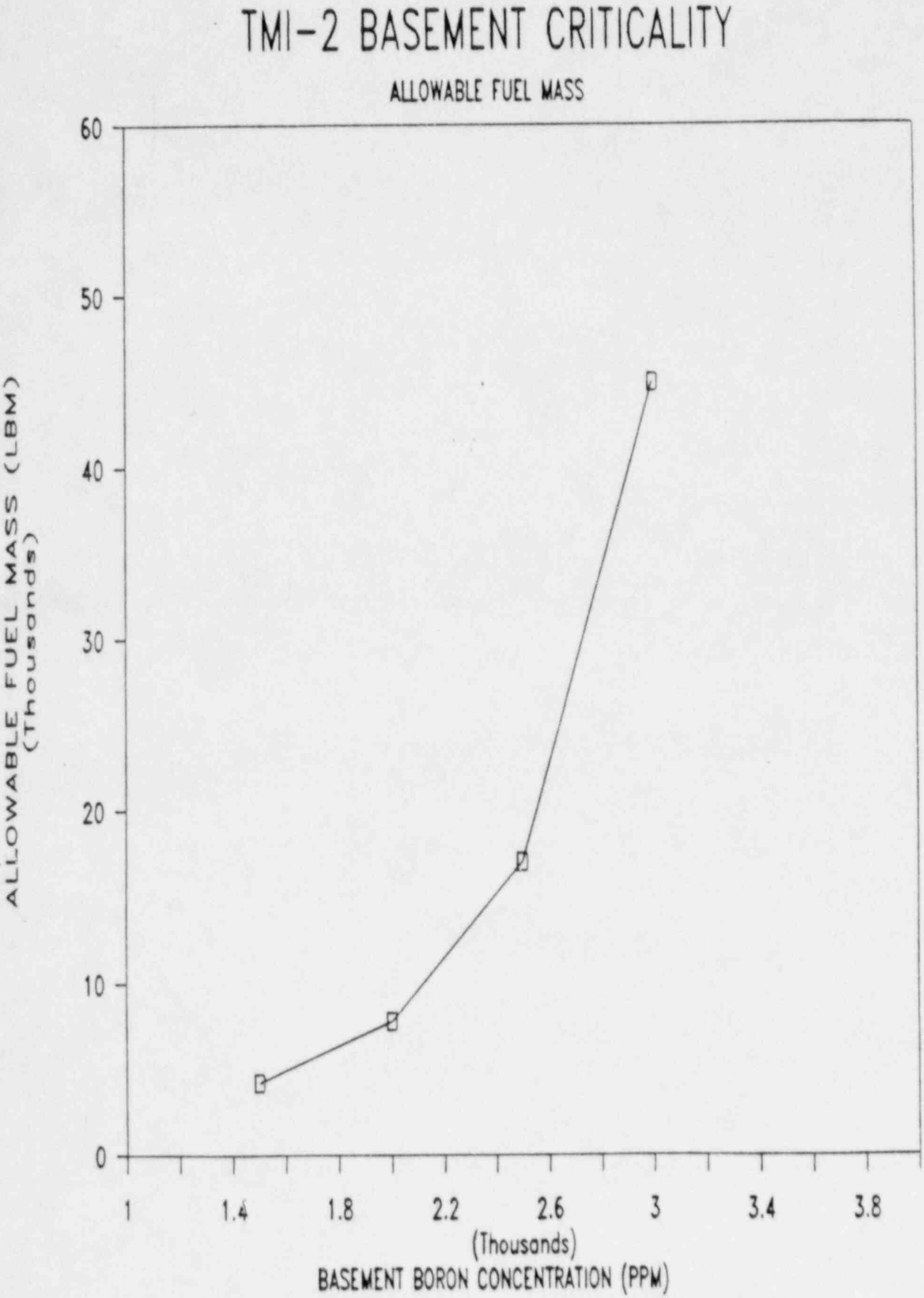

TMI-2 BASEMENT CRITICALITY
ALLOWABLE FUEL MASS
ALLOWABLE FUEL MASS (LBM)
(Thousands)
0
10
20
30
40
50
60
1
1.4
1.8
2.2
2.6
3
3.4
3.8
(Thousands)
BASEMENT BORON CONCENTRATION (PPM)

Figure 1