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September 15, 1988
4410-88-L-0138/0394P

US Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Dear Sirs:

Three Mile Island Nuclear Station, Unit 2 (TMI-2)
Operating License No. DPR-73
Docket No. 50-320
Safety Evaluation Report for the Completion of
Upper Support Core Assembly Defueling

Attached is the Safety Evaluation Report (SER) for Completion of Upper Core Support Assembly (UCSA) Defueling. The purpose of this SER is to demonstrate that the activities associated with cutting and moving the baffle plates to defuel the UCSA can be accomplished without jeopardizing the health and safety of the public.

The potential exists that some activities associated with this SER may be performed prior to the completion of Lower Core Support Assembly and Lower Head Defueling. Thus, the activities for which GPU Nuclear believes NRC prior approval is not required are specified in Section 1.2, "Scope," of this SER. A criticality safety assessment for use of the plasma arc torch to cut the UCSA baffle plates and core support shield has been submitted for NRC approval via GPU Nuclear letter 4410-88-L-0110 dated August 11, 1988.

Sincerely,

F. R. Standerfer
Director, TMI-2

RDW/emf

Attachment

cc: Senior Resident Inspector, TMI - R. J. Conte
Regional Administrator, Region 1 - W. T. Russell
Director, Plant Directorate IV - J. F. Stolz
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SAFETY ANALYSIS

SA No. 4710-3221-88-03

Rev. No. 0

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TITLE

SAFETY EVALUATION REPORT

FOR

COMPLETION OF UPPER CORE SUPPORT ASSEMBLY DEFUELING

Originator [signature] Date 8/22/88

CONCURRENCE

Lead Engineer [signature] Date 8/24/88 RG 8/26/88 SRG J. Bevelacqua per Telecon [signature] Date 9/7/88

RTR Robert J. Ryan Date 8/24/88 RAC 8/26/88

Design Cognizant Engineer Robert J. Ryan Date 8/24/88 RJR 8/26/88 Rad Con J.E. Tarpinian for D.W. Turner per telecon [signature] Date 9/7/88

APPROVAL

Mgr Eng. Section Paul J Kocbi Date 8/24/88 PJK 8/26/88 Site Ops Director [signature] Date [illegible] 9/8/88

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GPU Nuclear

No. 4710-3221-88-03

Title: Safety Evaluation Report for Completion of Upper Core Support Assembly Defueling

Rev.	SUMMARY OF CHANGE	Approval	Date
0	Initial submittal, GPU Nuclear letter 4410-88-L-0138.		9/88

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1.0 PURPOSE AND SCOPE

1.1 Purpose

The purpose of this Safety Evaluation Report (SER) is to demonstrate that the activities associated with cutting and moving the baffle plates (Figures 1 and 2) to defuel the upper core support assembly (UCSA) in the TMI-2 Reactor Vessel (RV) can be accomplished without jeopardizing the health and safety of the public.

1.2 Scope

This evaluation addresses the following activities:

- Cutting of the baffle plates for later removal.
- Removing bolts from the baffle plates.
- Removal of the baffle plates*, and
- Removal of core debris from the baffle plates and core former plates.

*Baffle plate structural material is intended to be stored in the RV or as an option in the core flood tanks.

It is noteworthy that a criticality safety assessment for cutting the baffle plates has been submitted for NRC review and approval (Reference 9). GPU Nuclear believes that the activities associated with removing bolts from the baffle plates (with the exception of one row of bolts which is necessary to maintain the plate in place); and removing the core debris from the baffle plates and core former plates are within the scope of References 1, 2, and 3. Therefore, these activities do not require prior NRC approval. The activities associated with the removal and transfer of the baffle plates, due to the potential load drop consequences, require prior NRC approval.

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

- vacuum system
- untorquing tool
- bolt drilling tool
- cavitating water jet
- Automatic Cutting Equipment System (ACES) including the plasma arc torch

- baffle plate transfer shield (transfer out of the RV only)

- equipment/tools as described in Reference 1

As the UCSA defueling operations proceed, the potential exists that activities or equipment described in this report or References 1, 2, and 3 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 References 1, 2 and 3 are created. If no such hazards or safety concerns are created, UCSA defueling may proceed based on the new or modified activities or equipment without a requirement to revise this SER; however, such changes would become part of the annual report required by 10 CFR 50.59, "Changes, Tests, and Experiments."

2.0 MAJOR ACTIVITIES AND EQUIPMENT

UCSA defueling will be performed in accordance with detailed approved procedures. Any of the approved activities performed or tools used during initial, core region and Lower Core Support Assembly/Lower Head (LCSA/LH) defueling are considered acceptable during UCSA defueling unless specifically precluded. The initial and core region defueling activities and tools are evaluated in Reference 1. Initial LCSA disassembly and defueling activities and tools are evaluated by GPU Nuclear in Reference 2 and reviewed by the NRC in Reference 4. LCSA/LH disassembly and defueling activities are evaluated in References 3 and 5 and are similar to baffle plate defueling activities. Operations to be performed during UCSA defueling include:

- Cutting the baffle plates within the RV

- Bolt removal from the baffle plates

- Structural material removal from the UCSA

- Defueling the baffle and former plates

- Structural material removal from the RV (optional)

- Replacing structural material in the UCSA (optional)

2.1 Activities

The current methods being considered for UCSA defueling include using the ACES cutting machine in conjunction with long-handled tools to dismantle the baffle plates to provide access to the former plates. Defueling activities will then use the cavijet and vacuum systems to effectively defuel the baffle and former plates. The methods being considered for disposition of the baffle plates to facilitate former plate defueling include moving the baffle plates within the vessel for temporary disposition while defueling or moving the baffle plates to the core flood tanks for temporary disposition while defueling.

Moving the baffle plates within the RV to permit former plate defueling includes temporarily relocating cut baffle plate sections within the vessel and returning the baffle plates to approximately original configuration within the vessel following core debris removal.

Moving the baffle plates to the core flood tanks to permit UCSA former plate defueling includes rigging and handling the baffle plates with a transfer shield from the RV to either core flood tank with the option of returning the baffle plates to the RV following UCSA defueling. In either case the potential exists for special handling of the burned through sections of the baffle plates.

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 References 1, 2, and 3.

2.2 Equipment

The baffle plate transfer shield is intended to be utilized when transferring baffle plate sections to the core flood tanks. The transfer shield will be fabricated from a 3-inch thick steel plate designed to accept baffle plate sections using an integral hoist. The transfer shield is designed to sit on top of the SWP T-slot and features a locking mechanism to prevent the drop of a baffle plate once it is loaded. The transfer shield is not required when the baffle plate evolution does not remove structural material from the RV.

Descriptions of other tools required for UCSA defueling were provided in References 1, 2, and 3.

3.0 COMPONENTS AND SYSTEMS AFFECTED

Other components or systems in addition to those described in References 1, 2, and 3 may be required to conduct the UCSA 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 References 1, 2, and 3.

4.0 SAFETY CONCERNS

4.1 General

An evaluation of the activities associated with UCSA defueling identified the following safety aspects:

- RCS Criticality Control
- Boron Dilution
- Hydrogen Evolution/Gas Generation
- Pyrophoricity
- Submerged Combustion
- Fire Protection
- Decay Heat Removal
- Release of Radioactivity
- RV Integrity
- Burning/Cutting Operations
- Heavy Load Drops
- Basement Criticality

Each of these issues is discussed below.

4.2 RCS Criticality Control

The evaluations provided by References 1, 2, 6, 9, and 12 generally bound this concern during UCSA defueling. Based on the results of these analyses, it is concluded that the plasma arc torch, with a maximum drainable coolant system inventory of 3.5 gallons of unborated water, can be used to dismantle the UCSA without developing a criticality safety concern within the RV.

The above conclusion is based on the operational limitations listed in References 4, 8, 9, 10, and 11.

4.3 Boron Dilution

Boron dilution concerns during UCSA defueling are bounded by the evaluations provided by References 1 and 12. To preclude the possibility of a hydraulic fluid leak leading to a possible critical configuration of fuel and moderator, all hydraulic fluid used with UCSA defueling tools will be borated to at least 4350 ppm boron (added as boric acid).

4.4 Hydrogen Evolution/Gas Generation

Generation of small quantities of hydrogen gas (less than 0.1 SCFM) will be a by-product of the plasma arc cutting tool operation underwater. This hydrogen will be diluted by the off-gas ventilation system, as required. Thus, a combustible concentration will not occur within the Reactor Building. The building purge system is then utilized to remove potentially toxic by-product gases produced during plasma arc torch operation. Other hydrogen related safety issues are bounded by the evaluations provided in Reference 1.

4.5 Pyrophoricity

Pyrophoricity concerns during UCSA defueling are bounded by evaluations provided in References 1 and 13.

4.6 Submerged Combustion

The use of the ACES plasma arc torch creates a heat source which was evaluated and reviewed in References 2 and 4. 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 14). It is considered reasonable not to postulate a combustion reaction of exposed fuel debris due to operation of the ACES plasma arc torch. Experience to date has confirmed that submerged combustion is not a concern during plasma arc torch operation.

4.7 Fire Protection

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

4.8 Decay Heat Removal

Decay heat removal concerns during UCSA 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 only raise the RCS temperature approximately 2°F. The RCS temperature will be monitored to preclude an unlikely uncontrolled water temperature increase. Experience to date has confirmed that water temperature is not measurably affected by operation of the plasma arc torch.

4.9 Release of Radioactivity

The central zone of the plasma arc gas reaches temperatures of 20,000°F to 50,000°F and is completely ionized. However, this high temperature 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 remain within the RV. Soluble isotopes trapped in the fuel matrix may become dissolved in the water. This possible increase in the concentration of radioactivity of RV water is not expected to be prohibitive or exceed that observed in the core drilling program.

Recent PCI cutting operations in the LCSA have caused minor changes in radiation levels through the off-gas ventilation system due to Krypton-85 gaseous releases. Defueling personnel are protected from direct/concentrated Krypton-85 releases via off-gas system operation during PCI cutting and releases outside the Reactor Building will be maintained within the limits given by Reference 1.

Other safety concerns associated with the release of radioactivity from the RV to the environment are bounded by the evaluations in Reference 1.

4.10 RV Integrity

RV Integrity concerns during UCSA defueling are generally bounded by the evaluation provided in Reference 3.

Based on the information from Reference 3, GPU Nuclear believes that UCSA defueling can be conducted without impairing the integrity of the RV.

4.11 Burning/Cutting Operations

Operation of burning devices inside the vessel has been evaluated by GPU Nuclear in References 2 and 10 which were reviewed by the NRC in Reference 4 and is also evaluated in Reference 3. UCSA burning/cutting operations are limited to inside the core support structure where the torch is at least 15 inches from the RV wall. Current cutting operations are expected to begin at the top of each of 8 or more locations on the baffle plates and cut the length of the plate. These cuts will be made after considerable experience is gained using the plasma arc torch elsewhere in the RV. The arc of flame of such burning devices, operating underwater, will always be operated at least 15 inches from the RV wall. Because of rapid dissipation of the arc energy propagation of an arc through the 15 inches (two 2 inch thick steel plates and 11 inches of water) is not possible. Thus, damage to the RV wall due to the operation of burning devices is precluded.

4.12 Heavy Load Drops

During UCSA defueling, the RV lower head and incore nozzles will be subject to potential direct load drops not previously evaluated. These potential load drops are discussed below.

The first and second potential load drops (i.e., manual tool positioner and manual tool positioner with assembled PCI mast and torch) are introduced during the baffle plate cutting evolution. The installation of the MTP, with only the PCI/MTP saddle plate assembly attached, through the T-slot represents a load drop of 3200 pounds from a distance of 47.6 feet. The MTP with the PCI mast and torch attached represents a potential load drop of approximately 4100 pounds from a distance of 23 feet. Calculations have demonstrated that these load drops will not result in a nozzle weld failure (Reference Appendix A - Cases C and D).

A third potential load drop occurs once the baffle plates are cut and bolts removed and the baffle plate section is ready to be moved. If the baffle plates are to remain in the vessel the largest baffle plate section is approximately 2500 pounds (including support clamps) and the drop distance is approximately 9.5 feet using a transfer lift of less than 2 feet. Calculations have demonstrated that this load drop will not result in a nozzle weld failure (See Appendix A, Case B).

The fourth and fifth potential load drops [i.e., baffle plate section (1/16th) and transfer shield] occur during the evolution option which removes the baffle plates from the RV for temporary disposition in one of the two core flood tanks. The most severe load drop potentials exist with the drop of a baffle plate section from its loaded position inside the transfer shield and the transfer shield drop onto the SWP. The baffle plate section (including support clamps) represents a load of approximately 1500 pounds dropped from Elevation 335'-0" or a distance of 44'-1" to the bottom head/incore nozzles. Calculations have demonstrated that this load drop will not result in nozzle weld failure (See Appendix A - Case A). The transfer shield will be designed so that it will not pass through the T-slot and therefore impacts the SWP on load drop. The weight of the transfer shield and baffle plate section will be approximately 24,000 pounds which could potentially be dropped on the SWP. The shield will be limited to a maximum lift height of Elevation 332.1 feet, based on Reference 16, which provides load lift limitations to prevent SWP collapse if a load is dropped anywhere on the platform. Reference 17 allows increased lift heights as long as the load path is limited to within 3'-6" of either side of the north-south or east-west centerlines of the SWP, according to the formula $\frac{34,120}{W} + 331.5$. With a load of 24,000 pounds the allowable lift height is 332.9 feet. For the remainder of the load path from the vessel to the core flood tank the restrictions per Reference 18 would apply.

The potential for other load drop accidents into the RV 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-3890.01, "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 potential for a criticality event in the Reactor Building basement was previously addressed in References 2 and 15.

The controls discussed in Section 4.13 of Reference 2 to ensure subcriticality of potential leakage into the cavity of the RV will continue to be maintained during UCSA defueling. Therefore, criticality is precluded. Monitoring of RB cavity will continue during UCSA defueling.

5.0 RADIOLOGICAL CONSIDERATIONS

Based on a comparison of activities associated with Reference 1 to those associated with UCSA defueling, it is concluded that the radiological considerations associated with UCSA defueling are bounded by Section 5 of Reference 1. However special precautions will be taken to prevent exposure of operating personnel during transport of radioactive and contaminated pieces of the UCSA within the RV to their storage location within the RV. Although these pieces of the UCSA will be inspected to ensure there is no visible fuel debris, all pieces are radioactive due to Co-60 activation and surface contamination by soluble fission products.

If the baffle plates are transferred to the core flood tanks utilizing the transfer shield, the plates will be visually inspected to assure that no fuel is present.

The measured radiation levels from the baffle plates reached 3000 R/h at a standoff distance of approximately 2 inches. If the baffle plates are to be removed from the RV, the 3 inches of steel on the transfer shield will limit the dose rate at a standoff distance of 50 feet to approximately 1 R/hr. Also, additional temporary shielding will be used as necessary to limit personnel exposures to considerably lower levels. The adequacy of personnel exposure control practices have been demonstrated by handling core debris in the core region and during LCSA plate handling without shielding of the plates.

Reference 5 estimated an occupational exposure to complete RV defueling of approximately 1580 person-rem. This estimate includes the evolution of moving the baffle plates to the core flood tanks (approximately 30 person-rem). The jobhours and person-rem expended to date for defueling activities are provided in Table 5.1.

TABLE 5.1

JOBHOURS AND PERSON-REM EXPENDED THROUGH MAY 31, 1988

Activity	Jobhours	Person-Rem
Preparation and Installation	5,120	120
Operation	43,534	423
Defueling Support	28,793	440
Maintenance	970	45
Decon and Removal*	0	0
Totals	78,417	1028

* No activity associated with final decontamination and removal of defueling equipment has currently been performed, 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 UCSA 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 UCSA 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, with 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 UCSA defueling. The analysis of these events provided in Section 4 results in the conclusion that the postulated events are bounded by previous evaluations and/or do not result in an unanalyzed condition.

To determine if UCSA 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. References 2 and 3 evaluated the consequences of potential events during LCSA/LH disassembly and defueling and demonstrated that LCSA/LH defueling can be performed safely.

The activities to be performed during UCSA defueling are bounded by the evaluations in References 1, 2, and 3 since UCSA defueling will be conducted within the heavy load limitations previously established in those references. Further, Reference 9 demonstrates that UCSA defueling does not pose a criticality safety concern.

By considering postulated events and reviewing various safety mechanisms (i.e., fire protection and decay heat removal), it has been demonstrated that UCSA 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 UCSA 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.

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

The variety of postulated events analyzed in References 1, 2, and 3 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 or other SERs previously approved by the NRC. Section 4 of this SER evaluates events postulated for UCSA defueling. These type of events have been previously evaluated and, therefore, do not represent a different type of accident or malfunction.

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

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 UCSA 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 RV to prevent overheating and potential criticality. Additional borated water has been added to the cavity beneath the RV to bring the boron concentration above 3500 ppm as specified in References 2 and 3. This action ensures that 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 because no more than 3.5 gallons of unborated water can be inadvertently drained into the RV (Reference 9).

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

In conclusion, the UCSA 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 UCSA 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 UCSA defueling activities can be performed with no significant environmental impact.

9.0 CONCLUSIONS

Activities associated with UCSA defueling have been described and evaluated. The evaluations have shown that the radioactivity releases to the environment that will result from the planned activities will not exceed allowable limits. It has been demonstrated 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 UCSA 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 Lower Core Support Assembly Defueling, Revision 2, 4710-3221-86-011, January 1988.

3. GPU Nuclear letter 4410-88-L-0060 dated June 6, 1988, "Safety Evaluation Report for Completion of Lower Core Support Assembly and Lower Head Defueling," Revision 0.

4. NRC Letter dated April 1, 1988, Lower Core Support Assembly Defueling.

5. GPU Nuclear letter 4410-88-L-0100 dated June 27, 1988, "Safety Evaluation Report for Completion of Lower Core Support Assembly and Lower Head Defueling," Revision 1.

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

7. 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.

8. GPU Nuclear letter 4410-88-L-0067 dated April 29, 1988, "Plasma Arc Torch Coolant System."

9. Criticality Safety Assessment for Use of the Plasma Arc Torch To Cut the Upper Core Support Assembly Baffle Plates and the Core Support Shield, 4710-3221-88-02, August 1988.

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

11. GPU Nuclear letter 4410-88-L-0026, dated February 26, 1988, "Response to NRC Comments on the Criticality Safety Assessment for Using the Plasma Arc Torch to Cut the LCSA."

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

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

14. EG&G Plasma Arc Test Report, LCSA-4, April 30, 1985.

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

16. Safety Evaluation Report for Heavy Load Handling Over the TMI-2 Reactor Vessel, Revision 0, 4350-3153-85-2, May 1985.

17. Safety Analysis for Lower Core Support Assembly Forging Removal From the Reactor Vessel and Lifting and Handling in the Reactor Building, Revision 0, SA No. 4710-3255-88-0022, July 1988.

18. Safety Evaluation Report for Heavy Load Handling Inside Containment and Over Fuel Pool "A", Revision 3, 4350-3153-85-1, June 1986.

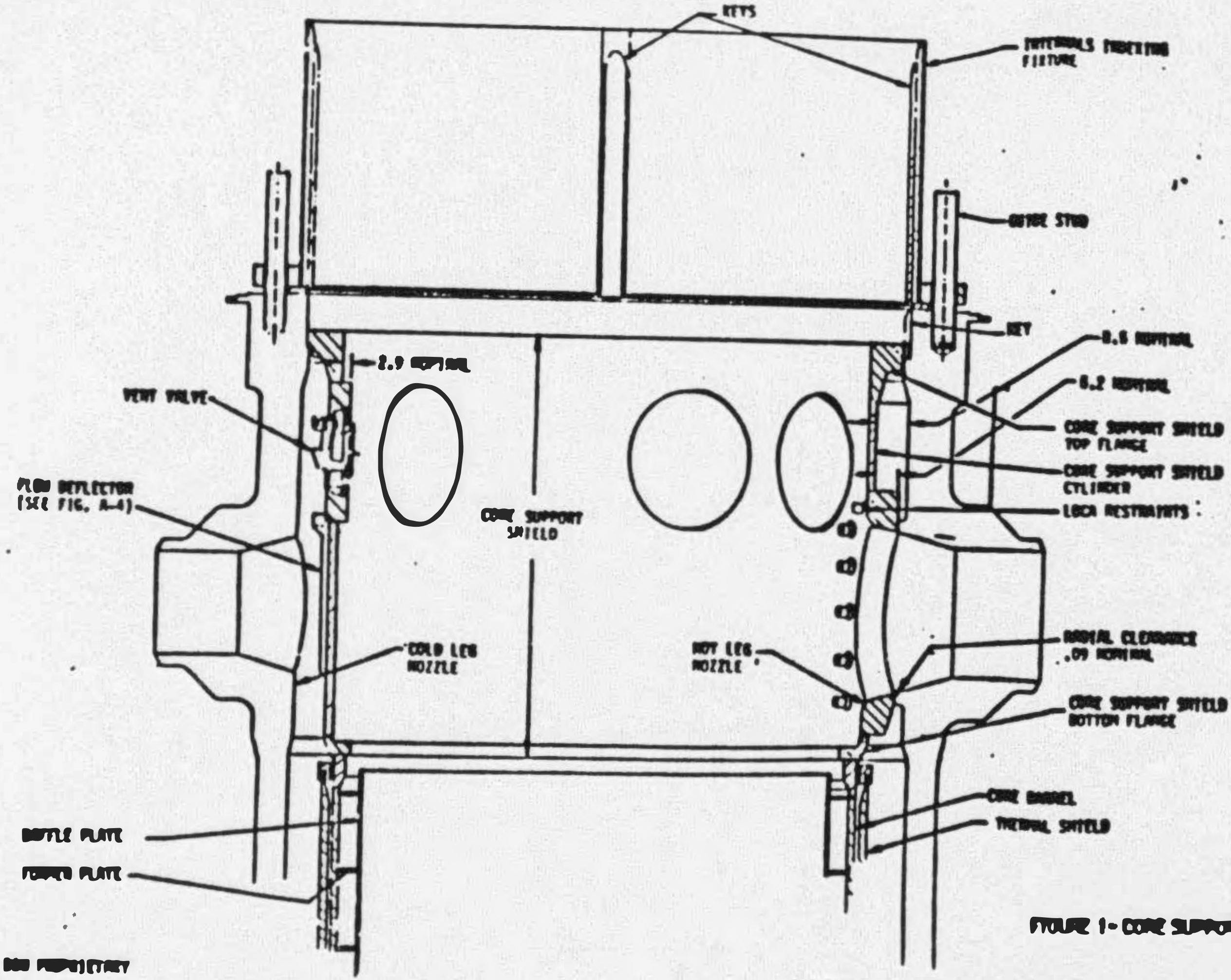

FIGURE 1- CORE SUPPORT ASSEMBLY DIAGRAM

B&W PROPRIETARY

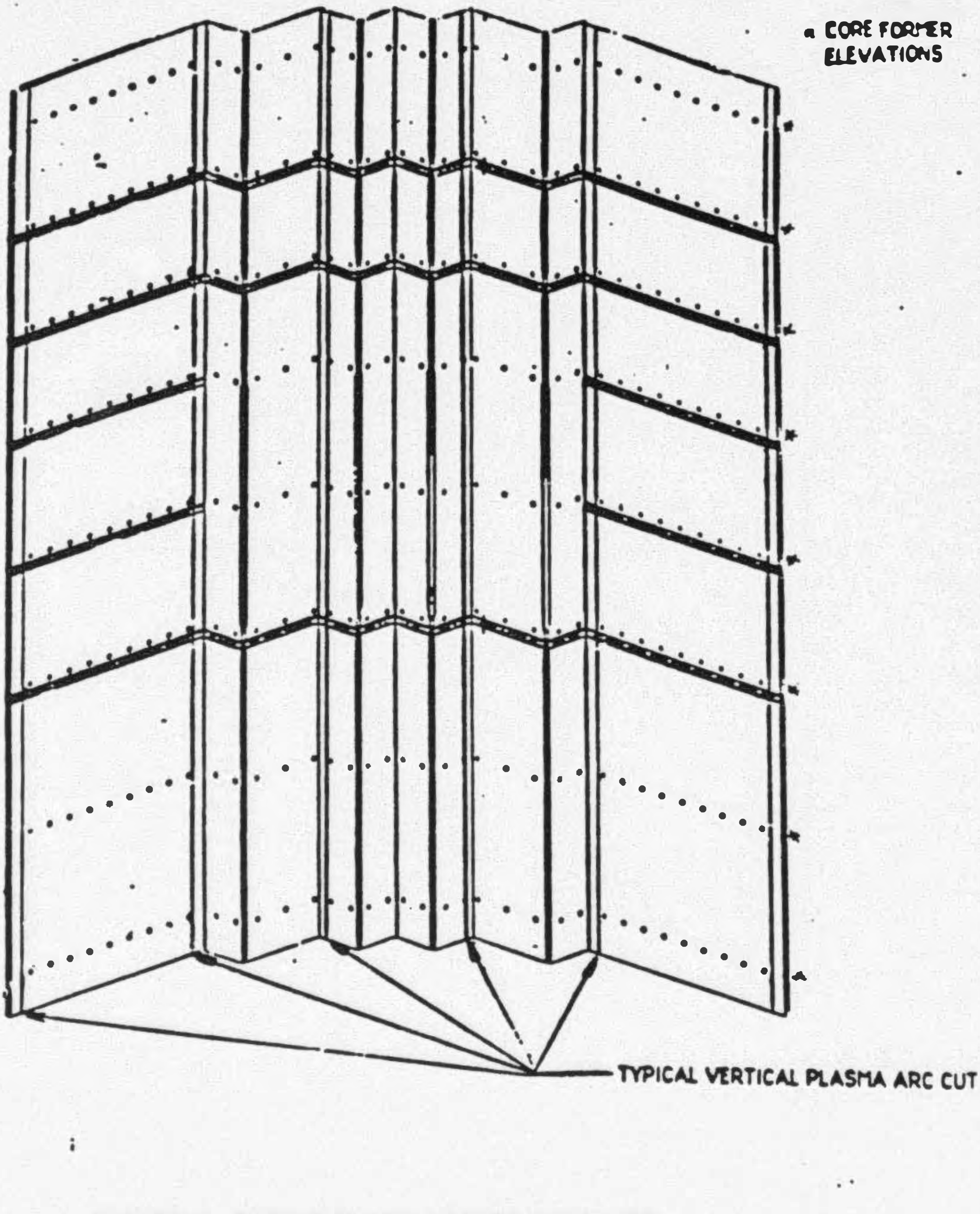

FIGURE 2- BAFFLE PLATE ACCESS OPENINGS

NOTE: HORIZONTAL BLACK BARS DO NOT REPRESENT ANY STRUCTURE, THEY ARE NECESSARY FOR THE CADD SYSTEM TO DEPICT HOLES IN A FLAT PLATE.

APPENDIX A

EVALUATIONS OF LOAD DROPS OVER THE REACTOR VESSEL

During upper core support assembly (UCSA) defueling, a large area of the RV lower head may be exposed to direct impact from heavy loads. Analyses have been performed to better determine the potential damage which could be incurred by the incore nozzles due to dropped loads. To provide the analyses reported herein, simple calculations were employed in order to ascertain if further, more complex analyses were warranted.

The following objects were considered as potential accidents loads:

A. Baffle plate section (1/16th section)

B. Baffle plate section (1/8th section)

C. Manual Tool Positioner

D. Manual Tool Positioner W/PCI Mast

In order to maintain a simplistic approach, the analyses made the following major assumptions:

1. Upon impact, all kinetic energy of the falling object is transmitted to the instrumentation nozzle and results in strain. This assumption is conservative since some of the energy would also be converted to strain in the dropped object and the RV lower head.

2. The compressive stress-strain curve for a short column of Inconel is identical to the tensile stress-strain curve. This assumption is conservative since ductile metals will fail in tension before failing in compression without buckling.

3. The static stress-strain curve for Inconel is appropriate for dynamic loadings. This assumption may be slightly unconservative as some metals exhibit higher strength but lower ductility with increasing load application speeds.

4. The strain is uniform over the entire nozzle. This assumption does not account for the possibility of the nozzle bending. (See page A-3.0 for bending considerations.) Use of this assumption gives an upper bound on the permissible drop heights.

5. As-constructed material properties were used for the nozzle and weld materials. However, nozzle material properties may have been degraded due to elevated temperatures during the course of the accident.

The objects under consideration, when dropped through water, will be subject to drag which could vary significantly, depending on the orientation of the falling object relative to the direction of movement. An examination of the potential coefficients of drag for various sharp edged bodies indicates drag coefficients varying from 0.5 to 1.5. This indicates that the drag coefficient will have a significant effect on the calculated impact velocity for a water drop height of 30 feet or more. Drag coefficients did not need to be considered for the drop heights of the baffle plates sections (1/8th and 1/16th) while drag coefficients equal to 1.16 (for rectangular saddle plate assembly) were used for the objects C & D.

Assuming that the impact load is entirely in the axial direction and along the centerline of the nozzle, an upper bound on the permissible drop heights can be established.

It is conservative to assume that all the kinetic energy of the impacting object must be absorbed in the nozzle. Since the nozzle's stress-strain curve is known, the limiting impact velocity can be determined. Knowing the maximum expected drop heights, a comparison of actual to allowable velocities was made.

The following drop heights were used based on the planned evolution involving the particular object and were found to be acceptable, i.e., will not result in incore nozzle weld failure.

TABLE A

Postulated Drop Heights

Object	Weight lbs.	Air Drop height-ft	Water Drop height-ft
A Baffle Plate Section (1/16th)	1500	7.5	36.6
B Baffle Plate Section (1/8th)	2500	N/A	9.5
C Manual Tool Positioner	3200	11.0	36.6
D Manual Tool Positioner w/PCI Mast	4100	N/A	23

All of the above analyses considered that the dropped tool struck the exposed incore nozzle on centerline. Realistically, the impacting object could strike the nozzle off-center creating both an axial load and a bending moment. An impact load on the nozzle taper would produce a lateral load and an additional moment would be created.

The magnitudes of the lateral load and bending moment are difficult to establish. However, by using the energy approach and simple inelastic equations for the deflection of an end-loaded cantilever beam, the maximum energy absorbed can be compared with that for the "axial load only" condition.

Analysis has determined that the nozzle is capable of absorbing a side load of about 6% of that which it can absorb as an axial load. If a substantial part of the postulated impact energy is applied horizontally, the nozzle is likely to fail. However, such failure would be expected to be above and parallel to the inside surface of the RV lower head. Therefore, nozzle failure due to off-center loading could fail the nozzle but not cause significant leakage since the in-vessel segment of the 3/4" schedule 160 Inconel pipe and its weld would likely remain.

The greatest load transmitted to the vessel would be for an axial impact load on the incore instrument nozzle. Since the nozzle outer diameter above the vessel wall (i.e., 2 inches) is greater than the RV penetration diameter (approximately 1 inch), the nozzle would have to shear through the vessel wall in order to punch a hole through the lower head. The ultimate axial stress capability of the nozzle is well below the ultimate strength of the vessel wall so that the nozzle will fail before the lower head is penetrated. An undamaged nozzle, therefore, cannot be pushed through the vessel wall.

Of the potential failure mechanisms, it is concluded that the worst case anticipated incore nozzle failure mechanism is shearing at the inside surface of the RV lower head.

As previously noted, the 3/4" schedule 160 portion of the instrument tube which penetrates the vessel wall is welded directly to the vessel wall. The 2" O.D. incore instrument nozzle is welded separately to the vessel wall and the 3/4" pipe. 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 0.005 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.

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. This leakage has previously been calculated to be approximately 0.40 gpm per nozzle penetration. Capability has been demonstrated (Reference GPU Nuclear letter 4410-84-L-0154 dated November 6, 1984, "Technical Specification Change Request 46") to provide make-up in excess of 17 gpm even in the event of a loss of off-site power.