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Document Control Desk
US Nuclear Regulatory Commission
Washington, DC 20555

Dear Sirs:

Three Mile Island Nuclear Station, Unit 2 (TMI-2)
Operating License No. DPR-73
Docket No. 50-320
Core Support Assembly and Lower Head Defueling

Attached for your review and approval is the Safety Evaluation Report (SER) for Core Support Assembly and Lower Head (CSA/LH) Defueling. The SER demonstrates that the proposed activities associated with CSA/LH defueling can be performed without presenting 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.

Sincerely,

F. R. Standerfer
Director, TMI-2

FRS/CJD/eml

Attachment

cc: Regional Administrator - Office of I&E, Dr. T. E. Murley
Director - TMI-2 Cleanup Project Directorate, Dr. W. D. Travers

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SAFETY ANALYSIS

SA # 4710-3221-86-011

Rev. # 0

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of 14

TITLE

SAFETY EVALUATION REPORT

FOR

CORE SUPPORT ASSEMBLY

AND

LOWER HEAD DEFUELING

Originator Robert J. Ryan Date 1/29/87

CONCURRENCE

Lead Engineer/RTR L H Porter Date 1/29/87 SRG [illegible] Date 2 Feb 87

Cognizant Eng. [illegible] Date 1/29/87 Rad Con D W Turner Date 1/30/87

APPROVAL

Mgr. Eng. Section R C Gallagher Date 1/29/87 Site Ops Director [illegible] Date 2/2/87

Rev.	SUMMARY OF CHANGE	Approval	Date
0	Issued for use.		2/87

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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 defueling the upper and lower core support assembly (CSA) and the lower head (LH) in the TMI-2 reactor vessel can be accomplished without causing unacceptable risk to the health and safety of the public.

1.2 Scope

This evaluation addresses the following activities:

- o Removal of core debris from upper and lower CSA, including removal of CSA structural material when such structural material and core debris are not readily separable.
- o Removal of CSA structural material to gain access to debris deposits within or below the CSA.
- o Removal of sections of the elliptical flow distributor from which core debris is not readily separable.
- o Removal of sections of elliptical flow distributor to gain access to debris deposits in the LH.
- o Removal of core debris from the LH.
- o Installation/operation/removal of additional equipment in support of the above activities.

NOTE: LH defueling by vacuuming was addressed in Reference 1.

CSA structural material not placed in defueling canisters may be stored in the reactor vessel or in other out of vessel temporary containers which will be addressed in separate documentation.

Additional equipment to that discussed in Reference 1 required to support these activities consists of:

- o cavitating water jet
- o plasma arc cutting tool
- o Automatic Cutting Equipment System (ACES)
- o robot manipulators

As the CSA/LH 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, CSA/LH 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

CSA/LH 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 CSA/LH defueling unless specifically precluded. The initial and core region defueling activities and tools are evaluated in Reference 1. Additional operations to be performed during CSA/LH defueling include:

- Core debris and structural material removal from the upper and lower CSA
- Cutting the upper and lower CSA within the reactor vessel
- Core debris removal from the reactor vessel LH

Descriptions of tools in addition to those described in Reference 1 to be used for CSA defueling are provided below.

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 4). Piping downstream of the pump is precluded from siphoning because it is fixed above the Reactor Coolant System (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.

Plasma Arc Torch

The plasma arc 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, nitrogen or CO_2, is used to aid in flushing away the molten metal from the cut and to provide insulation for the torch head. The maximum total gas flow will be less than 170 scfm. Instead of RCS grade water at 4350 ppm boron, the torch coolant system will be charged with 80% B-10 enriched boron to reduce the adverse conductivity effects of using RCS grade coolant. The volume of the system is approximately 5 gallons. 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.

Automated Cutting Equipment System

The Automated Cutting Equipment System (ACES) will position the plasma arc torch to cut the lower CSA structural elements to provide access to the fuel in the reactor vessel lower head. 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 lower CSA. The controller is located in a Command Center outside of the containment building and is supported by a computer-assisted-design model of the lower CSA. The operators are assisted with both video monitor and printer output.

Robotic Manipulator

Two hydraulic operated manipulator arms will be mounted on the Manual Tool Positioner (MTP) or other suitable masts. 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.

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.

3.0 COMPONENTS AND SYSTEM AFFECTED

Other components or systems in addition to those described in Reference 1 may be required to conduct the CSA/LH defueling activities. Where this is the case they will be the subject of separate correspondence.

4.0 SAFETY CONCERNS

4.1 General

An evaluation of the activities associated with CSA/LH defueling identified the following safety concerns:

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

Each of these issues are discussed below.

4.2 Criticality Control

The evaluations provided by References 1 and 11 bound this concern during CSA/LH defueling.

4.3 Boron Dilution

Boron dilution concerns during CSA/LH defueling are generally 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 CSA/LH defueling tools will be borated to at least 4350 ppm natural boric acid. The plasma arc torch requires water cooling to its tip to prevent overheating. In most applications, this fluid is demineralized water because it exhibits a very low conductivity. However, the conductivity of RCS fluid in this application is too high. Therefore, in order to provide a coolant with acceptable conductivity, the coolant will be borated with 80% B-10 enriched boron to approximately 800 ppm (Reference 15). Neutronically, this will provide the same criticality safety margin as 4350 ppm natural boron. Proper administrative controls will be used to control the boration of tools.

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 CSA/LH defueling are bounded by evaluations provided in References 1 and 14.

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 5 and 7). 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 CSA/LH defueling.

4.8 Decay Heat Removal

Decay heat removal concerns during CSA/LH 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 7.

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 has been considered. Initially, 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 CSA and will sequentially cut through the lower grid, lower grid flow distributor, lower grid forging, in-core instrument support plate to the elliptical flow distributor. Since torch access to the elliptical flow distributor is physically precluded by the CSA structure until the upper layers are removed, the elliptical flow distributor (which is more than one foot from the reactor vessel wall) will be cut with the plasma arc torch only after considerable experience is gained by its use elsewhere in the reactor vessel. 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 even when cutting the elliptical flow distributor.

Additionally, the use of other tools that could potentially impart excessive loads to the incore instrument tube nozzles or damage the reactor vessel wall (e.g., abrasive/water jet cutting system) 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 CSA has been removed after which procedural limitations will be applied. Mechanical cutting devices, such as the abrasive saw, grinding wheel, cavitating water jet 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 further.

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 6.

4.12 Heavy Load Drops

After a portion of the lower flow distributor has been removed, the incore instrument nozzles and the reactor vessel lower head will be exposed to the potential of impact by dropped loads. Prior to that time, the CSA structure will preclude the dropping of heavy loads on the incore nozzles.

Appendix A describes analyses which demonstrate that load drop characteristics associated with LH defueling do not significantly compromise the integrity of the reactor vessel after the elliptical flow distributor is removed or sectioned. In addition, the consequences of the total failure of an incore nozzle have been previously evaluated. It was concluded that GPU Nuclear has the capability to promptly detect a totally failed nozzle penetration and can maintain the RCS level at or above the reactor vessel nozzles (References 6 and 16).

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. 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.

5.0 RADIOLOGICAL CONSIDERATIONS

Based on a comparison of activities associated with Reference 1 to those associated with CSA/LH defueling, it is concluded that the radiological considerations associated with CSA/LH defueling are bounded by Section 5 of Reference 1. 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 December 1986

Activity	Jobhours	Person-rem
Preparations, installations	3,930	100
Operations	18,171	188
Maintenance/Support	9,693	151
Decontamination and Removal*	0	0
TOTALS	31,794	439

*No activity associated with final decontamination and removal of defueling equipment has been performed as of January 1, 1987, 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 CSA/LH 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 CSA/LH 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 CSA/LH 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 CSA/LH defueling activities involve an unreviewed safety question, the following questions must be evaluated.

<u>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?</u>

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 CSA/LH defueling are bounded by previous evaluations.

The proposed activities for CSA/LH defueling will eventually create a hole in the lower CSA, exposing a large area of the lower RV to direct impact from a heavy load. Appendix A provides a summary analysis of the potential damage to the reactor vessel LH incurred by a heavy load drop. This analysis concludes that the potential load drop of either a loaded canister in sleeve or the Manual Tool Positioner and Manipulator (MTP/M) directly on an exposed incore instrument nozzle may result in leakage of the RCS. However, the MTP/M, as discussed in Appendix A, will only be in position to cause damage when it is being taken in or out of the reactor vessel. The majority of the time, the MTP/M will be in position within the reactor vessel where the drop height is minimal. The canister sleeve handling tool and the Canister Positioning System (CPS) both have locking devices to prevent dropping of a loaded canister and sleeve. The locking device on the canister sleeve handling tool is verified to be engaged prior to lifting the canister and sleeve. The locking device on the CPS is verified to be engaged after the canister sleeve is positioned on the CPS. In addition, previous evaluations have shown that sufficient leak detection and mitigation equipment is available and operable should a load drop event, which damages the reactor vessel LH, occur.

The design features and administrative controls as described in Reference 9 ensure that the probability of a load drop is minimized. GPU Nuclear will also take special precautions for the handling of the MTP/M during installation/removal from the reactor vessel.

By considering postulated events and reviewing various safety mechanisms, i.e., fire protection and decay heat removal, it has been demonstrated that CSA/LH defueling activities will not adversely affect 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 CSA/LH defueling do not increase the probability of occurrence or the consequences of any remaining 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 demonstrates that the potential events postulated for CSA/LH defueling are bounded by previous evaluations and do not create the possibility of occurrence of an accident or malfunction of a different type than evaluated previously in the safety analysis report.

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 CSA/LH defueling process. Subcriticality is ensured by establishing the 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.

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

In conclusion, the CSA/LH 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 CSA/LH 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 CSA/LH defueling activities can be performed with no significant environmental impact.

9.0 CONCLUSIONS

Activities associated with CSA/LH 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 CSA/LH 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. Technical Evaluation Report for Defueling Canisters, Revision 2, 15737-2-G03-114, January 1986.

3. Safety Evaluation Report for Canister Handling and Preparation for Shipment, Revision 3, 15737-2-G07-111, June 1986.

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

5. EG&G Plasma Arc Test Report, LCSD-34, April 30, 1986.

6. GPU Nuclear letter to W.D. Travers, USNRC, 4410-86-L-0162 dated September 19, 1986, Subject: Core Bore Operations, and Attachments.

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

8. Technical Specification Change No. 46

9. Safety Evaluation Report for Heavy Load Handling over the TMI-2 Reactor Vessel, Revision 0, 15737--2-G07-110, April 18, 1985.

10. GPU Nuclear letter to W. D. Travers, USNRC, 4410-86-L-0160 dated September 9, 1986, Subject: End Fitting Storage.

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

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

13. TMI-2 Reactor Building Purge - Kr-85 Venting, GEND-013.

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

15. Addition of Soluble and Insoluble Neutron Absorbers to the Reactor Coolant System of TMI-2, GEND-026 dated July 11, 1986.

16. Extended Core Stratification Sample Acquisition Activity, GPU Nuclear letter 4410-86-L-0122 dated July 11, 1986.

APPENDIX A

EVALUATIONS OF LOAD DROPS OVER THE REACTOR VESSEL

During core support assembly and lower head defueling, the lower core support assembly (CSA) will have pieces cut from it and removed to gain access to core debris. Eventually, a hole will be created through the lower CSA, exposing a large area of the lower reactor vessel head 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 accident loads:

TABLE A

Maximum Achievable Drop Heights for Considered Objects

OBJECT	DROP DISTANCE IN AIR**	DROP DISTANCE IN WATER*
A. Light Duty Pole	52'-0"	36'-7"
B. End Effector Handling Tool	56'-0"	36'-7"
C. Loaded Defueling Canister	5'-6"	36'-7"
D. Loaded Defueling Canister in Sleeve	N/A	24'-0"
E. Manual Tool Positioner w/Manipulator	21'-7"	36'-7"

*Distance to bottom, inside surface of Reactor Vessel Lower Head
**Drops are sequential - first air then water

The analyses, in order to maintain a simplistic approach, 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 lower reactor vessel 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 they do 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 is not a conservative assumption as the real possibility of the nozzle bending is neglected (see page 3 for bending considerations). Use of this assumption gives an upper bound on the permissible drop heights.

5. Virgin material properties were used for the nozzle and weld materials. This assumption is slightly non-conservative since the nozzle material properties at 1200°F have not been determined.

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 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. In lieu of actually calculating drag coefficients for all dropped objects, a range of drag coefficient from 0.5 to 1.5 was used.

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 impact velocity allows the determination of the drop heights by iteration.

The following drop heights were calculated.

TABLE B

Allowable Drop Heights

Object	Weight lbs.	Cross Sectional area-in.^{2}	Maximum Strike Velocity-in/sec	Air Drop height-ft 0.5	Air Drop height-ft 1.5	Water Drop height-ft 0.5	Water Drop height-ft 1.5
A	150	2.8	2120	>52.0	>52.0	36.6	36.6
B	500	9.6	1160	>56.0	>56.0	36.6	36.6
C	3350	154	449	--	> 5.5	34.1	36.6
D	5100	254	364	--	--	19.6	>24
E	4500	128	388	--	--	21.7	25.4

A comparison of the calculated drop heights versus the criteria previously given in Table A shows that even for the very low drag coefficient (0.5) items A & B (the Light Duty Pole and the End Effector Handling Tool) satisfy the given criteria. The loaded defueling canister with the minimum drag coefficient misses the water drop height criteria by about two feet (34.1' vs.

36.6') and the loaded defueling canister with sleeve misses by about four feet (19.6' vs. 24'). Note that with the maximum drag coefficient of 1.5 both criteria are met. The manual tool positioner does not attain the maximum drop height criteria by several feet with either drag coefficient.

A more realistic evaluation of the criteria for the dropped fuel canister indicates that the loaded canister when in a "droppable" position is a) within the Canister Positioning System (CPS) sleeve or b) within the port of the shielded work platform or c) over the port in the shielded work platform. For each of the positions from which it might drop, it would strike the CPS first thereby decreasing its velocity. Further, the criteria of all the impact energy being transmitted to the incore nozzle is highly conservative relative to the fuel canister; a vessel with a 1/4" thick shell. In all likelihood dropping the fuel canister on end onto the incore nozzle will result in significant bending and possibly puncture of the bottom head of the defueling canister and little or no deflection of the incore nozzle. Consequently, only the loaded canister in sleeve and the Manual Tool Positioner and Manipulator do not satisfy the drop criteria. The canister sleeve handling tool and the CPS both have locking devices to prevent dropping of a loaded canister and sleeve. The locking device on the canister sleeve handling tool is verified to be engaged prior to lifting the canister and sleeve. The locking device on the CPS is verified to be engaged after the canister sleeve is positioned on the CPS. Additionally, the dropping of a loaded canister and sleeve can only occur during a transfer of the sleeve from a loading position to the top position on the CPS. Consequently, the loaded canister and sleeve have a very low probability of dropping.

The Manual Tool Positioner will be in a position where it is more than 22 feet above the reactor vessel lower head less than one percent of the time it is in the reactor vessel. Obviously, most of the time this tool is in the vessel after holes have been cut through the elliptical flow distributor is when it is being used to perform work on either the lower CSA or the RV bottom head. The drop height from these positions is minimal. Further, when the tool post is fully retracted it is at elevation 313'-6" or approximately 22 feet above the lower head. In this position the tool is supported on its rails and not on the lifting rig. Consequently, the tool has a very low probability of dropping.

All of the above analyses considered that the dropped tool struck the exposed incore nozzle on centerline. A realistic condition exists whereby the impacting object strikes 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 again 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 as a side load only 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 reactor vessel 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 potential of punching a hole through the lower head is greatest for an axial impact load on the incore instrument nozzle. As a worst case it was assumed that the ultimate axial load on the nozzle had to be taken in direct shear of the lower reactor vessel head shell. The stresses determined are well below the ultimate strength of the vessel wall. An undamaged nozzle, therefore, cannot be pushed through the vessel wall.

Of the potential failure mechanisms, it is concluded that the worst anticipated incore nozzle failure mechanism is shearing off the nozzle at the inside surface of the reactor vessel 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 0.010 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.

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.

This analysis indicates that a potential load drop directly onto an exposed incore instrument nozzle may result in the leakage of reactor coolant water through the nozzle - vessel hole annulus. Previous submittals have shown that sufficient leak detection and mitigation equipment is available and operable to combat leakages due to the discharge of an entire incore nozzle (125 gpm) from the reactor vessel. Consequently, the capability exists to promptly detect the existence of any failed incore instrument nozzle and to maintain the reactor coolant system water level at or above the reactor vessel nozzles.