SAFETY ANALYSIS

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TITLE

SAFETY EVALUATION REPORT

FOR

COMPLETION OF LOWER CORE SUPPORT ASSEMBLY

AND LOWER HEAD DEFUELING

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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 removal of the gusseted incore guide tubes and the elliptical flow distributor; and defueling of the lower head (LH) 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:

- Removal of core debris from within the Lower Core Support Assembly (LCSA), including removal of LCSA structural material.
- Removal of the gusseted incore guide tubes and sections of the elliptical flow distributor including sections from which core debris is not readily separable to gain access to debris deposits in the lower head.
- Removal of 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 RV, stored out of the RV in temporary containers, or stored in the Reactor Building (e.g., Reactor Building Basement, Core Flood Tanks).

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

- core bore machine
- cavitating water jet
- Automatic Cutting Equipment System (ACES) including the plasma arc torch
- equipment/tools as described in Reference 1

*NOTE: The elliptical flow distributor is considered to be part of the LCSA.

As the LCSA/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, LCSA/LH 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

LCSA/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 LCSA/LH 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 3. Operations to be performed during LCSA/LH defueling include:

- o Cutting the LCSA within the RV
- o Core debris and structural material removal from the LCSA
- o Core debris removal from the RV lower head

2.1 Activities

As discussed in References 2 and 3, the current method of dismantling and defueling the LCSA will utilize a core bore machine in conjunction with the ACES. The dismantling of the LCSA will also provide better access to the lower head for defueling.

Various combinations of the use of the core bore machine and the ACES may be utilized to obtain the most effective use of each. However, operations will most likely be similar to the following. Use the core bore machine to sever the gusseted incore guide tubes from the LCSA. The exact number, location, and the amount of boring to be performed on each piece may vary as operations progress. After these operations are completed, use the ACES to complete severing any remaining incore guide tubes and to cut the elliptical flow distributor and any remaining LCSA sections that require cutting.

After the respective boring and cutting operations, the severed material will be flushed and removed from the RV. The gusseted incore guide tubes and possibly other small pieces of the LCSA will be removed from the RV in fuel canisters or in special storage containers based on an inspection of the material. The sections of the elliptical flow distributor will be lifted, flushed, inspected to ensure no visible fuel is present, removed from the RV, and stored in a suitable Reactor Building location.

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/LH defueling were provided in References 1 and 2.

3.0 COMPONENTS AND SYSTEMS AFFECTED

Other components or systems in addition to those described in Reference 1 may be required to conduct the LCSA/LH 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/LH defueling identified the following safety aspects:

- 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
- Basement Criticality

Each of these issues is discussed below.

4.2 RCS Criticality Control

The evaluations provided by References 1, 4, 5, and 7 generally bound this concern during LCSA/LH defueling. Based on the results of these analyses, 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, including the elliptical flow distributor head, without developing a criticality safety concern within the RV.

The above conclusion is based on the operational limitations listed in References 3, 6, 7, and 8.

4.3 Boron Dilution

Boron dilution concerns during LCSA/LH defueling are bounded by the evaluations provided by References 1 and 9. 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/LH defueling tools with the exception of the core bore machine will be borated to at least 4350 ppm boron (added as 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 10).

4.4 Hydrogen Evolution

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 treatment system, as required. 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/LH defueling are bounded by evaluations provided in References 1 and 11.

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 3. 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 12). It is considered reasonable not to postulate a combustion reaction of exposed fuel debris due to operation of the ACES plasma arc torch.

4.7 Fire Protection

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

4.8 Decay Heat Removal

Decay heat removal concerns during LCSA/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 only raise the RCS temperature approximately 2°F. The RCS temperature will be monitored to preclude an unlikely uncontrolled water temperature increase.

4.9 Instrument Interference

Operation of the plasma arc torch within the RV, to date, has not resulted in any disruption of any Technical Specification required instrumentation.

4.10 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 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 RV to the environment are bounded by the evaluations in Reference 1.

4.11 RV Integrity

The following subsections demonstrate that LCSA/LH defueling has a low probability of impairing the integrity of the RV. Section 4.11.1 discusses RV incore nozzle integrity based on the thermal hydraulic evaluations performed in References 13 and 14. These evaluations demonstrate that the temperature required for deformation of the RV lower head is well below the melting temperature for the incore nozzle welds. Since no evidence of RV leakage has been observed during or after the accident, RV failure did not occur and consequently the inside surface of the lower head clad did not reach melting. Thus, GPU Nuclear concludes that there is a high probability that the incore nozzle welds have maintained their original integrity. Sections 4.11.2 through 4.11.4 further discuss the RV incore nozzle integrity and demonstrate that based on visual examination of incore nozzles, verification of nozzles based on measured thermocouple junction cable lengths, and verification of nozzle integrity based on stub assembly removal that no information which contradicts the conclusion of Section 4.11.1 has been identified. The information in Sections 4.11.1 through 4.11.4 is summarized in Section 4.11.5.

It is noteworthy that none of the evaluations in Sections 4.11.1 through 4.11.5, taken separately, provide complete confirmation of nozzle integrity because of the inaccessibility of some of the nozzles. However, when taken together, the absence of negative findings and the variety of positive indication provide a preponderance of evidence to support a conclusion that the nozzle welds are undamaged and, therefore, the lower RV head integrity is sufficient to withstand worst case bounded heavy load drops.

4.11.1 Verification Of Incore Nozzle Integrity Based On Thermal Hydraulic Evaluation

As discussed in Reference 13 increasing metal temperatures will decrease metal strength. At 1600°F, the ultimate strength of the RV lower head carbon steel material is only 13,400 psi. At a 2200 psi, internal pressure and a metal average temperature of 1600°F, the 5-inch thick lower head would either rupture or experience significant plastic deformation. Pressure transients to approximately 2200 psi were experienced in the RV after molten core material relocated into the lower head. Since the RV lower head has not experienced post-accident leakage, it must be concluded that the RV lower head carbon steel shell did not creep and, therefore, did not attain average temperatures of 1600°F.

The conservative thermal hydraulic evaluation contained in Reference 13, demonstrated that if the temperature of a significant portion of the lower head reached or exceeded temperatures of 1600°F, the incore nozzle welds would not have reached their melting temperature of 2760°F. It also was noted that the upper part of the incore instrument nozzle could reach temperatures well above its melting point if surrounded by hot corium and with no cooling except for conduction down the nozzle and into the lower head. However, the half-inch or so of the nozzle above the inner surface of the lower head was close enough to the head that conduction of heat into the head provided sufficient cooling to keep this welded portion of the nozzle from melting.

Reference 14 predicts the temperature history of selected locations of the RV lower head for six (6) cases of material composition of the core rubble. These cases assume different thermal response characteristics of solid fuel material, fuel debris (fuel and clad) and control rod material in the lower head. A portion of the conclusions are repeated below:

> "The thermal response of the TMI-2 lower RV has been analyzed for three assumed lower plenum degraded core material configurations, [i.e., (a) a porous debris bed resting on the vessel head, (b) a debris bed resting on top of approximately eight (8) inches of consolidated molten fuel adjacent to the vessel, and (c) a porous debris bed resting on top of approximately eight (8) inches of assumed control rod material adjacent to the vessel]. For each configuration, the vessel thermal response was calculated assuming the debris was both coolable and noncoolable.
>
> "The calculations show a wide range of vessel thermal response is possible based on the debris configuration and debris cooling assumptions. Vessel melting temperatures were predicted for two (2) of the cases (Cases 3 and 4); however, for the relatively short transient (5400 seconds) very little melting was predicted. The most rapid heatup (resulting in the highest vessel wall temperatures) occurred for the case with assumed consolidated fuel adjacent to the vessel wall. For this case, temperatures in excess of 1100°K (1521°F) were achieved in less than 20 minutes and these temperatures are expected to have resulted in creep rupture during the first hour after the major core relocation. Cooling of the porous debris resting on top of the consolidated molten material had little effect on the maximum vessel temperatures for this case.
>
> "The calculations show for a porous debris bed that vessel wall temperatures would have been sufficiently low that creep rupture of the vessel would not be expected. In addition, a layer of control rod material adjacent to the vessel wall does provide an effective insulation to the wall at locations away from the wall/fuel debris interface."

Although the above conclusions show that for the two "worst case" assumptions vessel melting is predicted, the report also clearly shows that vessel rupture due to creep will occur at vessel average wall temperatures well below melting. As an example, at an average wall temperature of 1050°K (1430°F) and a pressure of 10.0 MPA (1450 psia) vessel rupture due to creep would occur in approximately 10 minutes. The analysis in Reference 13 predicted a maximum temperature difference between inside vessel surface temperature and average wall temperature of 435°F. Assuming this analysis to be correct, vessel rupture due to creep failure should have occurred at RV lower head surface temperatures of less than 2000°F; 650°F less than clad melting temperatures. GPU Nuclear, therefore, concludes that since no evidence of RV leakage has been observed during or after the accident, RV failure did not occur and consequently the inside surface of the lower head clad did not reach melting.

Gamma scan data (Reference 15) indicates the possible presence of a non-fuel layer of material in the RV lower head. This layer is expected to be approximately nine (9) inches in height at the centerline of the RV. The material could be either control rod material (silver, indium, and cadium) or control rod material mixed with stainless steel. The presence of this low melting point material would have protected the lower RV vessel welds from melting from hot corium located above the layer of melted high density material. This is also stated in the conclusion of Reference 14.

Reference 16 includes a study of the lower head thermal response to contact with a jet of melted core debris falling to the lower head in a period of 75 seconds (depicted by nuclear instrumentation response during the TMI-2 accident). The conclusions relating to the lower head analysis are repeated below:

> "Thermal damage potential to the lower head was also assessed for the configuration of coherent jet impingement of relocating melt debris. For this assumed configuration, the thermal response of the lower head is largely dictated by the contact time and heat transfer characteristics at the jet impingement surface. Assuming a jet diameter equal to the flow area within a single undegraded fuel assembly, the time for melt relocation as a jet is estimated to be about 75 seconds. This estimate is consistent with source-range monitor data, indicating that major core relocation occurred over a one (1) minute period.
>
> "Two (2) limiting conditions were assumed with respect to jet-impingement heat-transfer characteristics. The first was for a weak jet with conduction-limited heat transfer. The second was for strong jet forces where turbulent mixing and mass transfer effects at the impact surface lead to enhanced convection-controlled heat transfer process. For conduction-controlled heat transfer, surface ablation of the lower head by direct impingement is not inferred. This is due to the rather poor conductivity of the molten ceramic material and the high thermal capacity of the vessel head, which serves as an efficient and quick-response heat sink. However, calculational results for convection-controlled heat transfer indicate limited melt ablation at the liner

surface. The calculated depth of penetration of the melt front is on the order of about 0.5 inches (versus a head thickness of 5.5 inches) for a jet impingement time on the order of 75 seconds. A direct jet impingement time of about 15-20 minutes is, however, calculated to be necessary for melt ablation of the vessel head 1/2 thickness. It is, therefore, concluded that for a jet impingement time of 1 to 2 minutes (time associated with melt drainage to the lower plenum), little thermal damage to the lower vessel head would result."

Visual examinations in the RV and lower head indicate that the possibility of continuous jet impingement and consequent damage to an incore nozzle is extremely remote since:

- Each of the 175 fuel stub assemblies removed from the RV had some intact zirconium fuel tubes still attached and had a semi-intact lower end fitting. Only in core locations R-6 and R-7 was it evident that significant amounts of molten corium may have passed through the space to the lower internals. The balance of the major core relocation to the lower internal area was most likely outside the nozzle area.

- The large hole in the core baffle above grid positions R-6/R-7 lends credence to the suspected corium flow path to the lower head through the core formers and, thus, outside the area of most of the active core.

- The incore nozzle in position R-7, assembly No. 45, was observed to be standing but damaged at its upper end. This nozzle location, along with nozzle locations P-6, O-5, N-4, and M-3, are believed to have been in the flowpath of molten corium from the large hole in the baffle plates on the east side of the RV to the lower head. Resolidified material in R-6 and R-7 tends to confirm this flow path. Thus, based on the analysis in this section and Section 4.11.2, GPU Nuclear believes that these nozzle welds are intact.

Based on the above analysis, it can be concluded that the molten corium did not melt the lower head even if it were a molten jet which flowed into the lower head in only one location. Therefore, since the RV lower head did not melt it is reasonable to conclude that the incore nozzle welds have maintained their integrity.

4.11.2 Visual Examination of Incore Nozzles

GPU Nuclear received NRC approval (References 17 and 18) to disassemble and remove 15 of 52 incore instrument guide tubes. This approval was based on the video inspection of incore nozzles which revealed no observable damage of the incore nozzle to RV weld. Currently, 12 nozzles have been visually examined.

It has previously been evaluated by GPU Nuclear (References 17 and 19) and reviewed by the NRC (Reference 20) that significant incore nozzle weld damage would be highly unlikely if the nozzle exists above the vessel wall. In the video examination in the lower head region (Reference 21), 12 nozzles or portions of nozzles were seen. These included K-12, H-13, G-13, F-13, F-12, R-7, R-10, O-12, M-14, L-13, D-14, and C-13. Other than the limited damage to the tip of R-7 all other visible portions of these nozzles were undamaged. For five (5) of these nozzles, portions of the weld were seen; all these were also intact.

In the same video inspections, the incore instrument guide tubes were observed at 39 locations (including 12 where the nozzle was seen). For all of these locations, the visible portion of the guide tube was undamaged, except for thermal damage to the D-10 and R-7 guide tubes. Furthermore, even for the locations which show limited thermal damage, GPU Nuclear believes that the nozzle welds are intact. The thermohydraulic analysis in Section 4.11.1 indicates that even if the nozzles were immersed in molten corium, the original weld integrity would be maintined. Accordingly, GPU Nuclear believes that the video evidence in the lower head shows no indication that any of the nozzle welds have been damaged.

4.11.3 Evaluation Of Incore Nozzle Integrity Based On Thermocouple Junction and Self Powered Neutron Detector Test Data.

A study performed by EG&G (Reference 22) obtained accurate loop resistance measurements of the incore thermocouples including the extension cabling. With these data, it was possible to determine the actual loop resistance of the incore thermocouples following the accident. By comparing the post installation data with the present data, it was possible to identify changes in the length of the thermocouples as a result of the accident. The data shows that of the closed loops measured, all of these measured lengths of the thermocouple junctions are above or at the elevation of the cladding of the lower RV head. The thermocouple lengths in Table 1 were measured from the Tangent of the cladding of the RV lower head.

GPU Nuclear believes that if the thermocouple juncture is still operative and at a location above the inside of the vessel wall, it is highly unlikely that the entire incore nozzle had melted to a location below the juncture. As shown on Table 1, eleven (11) of the 52 thermocouples exhibited an open circuit and no location of the juncture was observed.

The junction locations of the 41 measurable thermocouples ranged from location E-11 which indicates a junction at the clad wall elevation to D-14 where the junction appeared to be 4-1/2 feet below its original location above the top of the core. Thus, the information provided in Reference 22 provides additional support to the conclusion that 41 incore guide tube nozzles have sound (i.e., undamaged) welds joining them to the RV lower head.

Reference 22 also provides results of in situ Self Powered Neutron Detector (SPND) test data for the incore assemblies, including the eleven (11) incore locations with open thermocouple junction data. For three (3) locations (i.e., H-13, O-6, and L-13), the in situ test results indicate that there was at least one (1) "good" SPND in the lower level of the core. This implies that the incore was intact up to an elevation corresponding to the lower portion of the core. For another seven (7) incore grid locations (i.e., K-12, F-13, E-4, F-3, M-3, P-6, and O-10), all seven (7) SPNDs were found to have an open circuit. Since an operating incore detector exhibits an open circuit by design, the open circuit conditions are indicative of less severe failure than are shorted conditions. Therefore, these locations also represent intact incore strings. The remaining incore location (i.e. G-11) had both open and shorted SPNDs.

4.11.4 Verification Of Incore Nozzle Integrity Based On Stub Assembly Removal

During removal of stub assemblies, trailing incore strings were observed in a number of locations and in other locations extensions of the incore string were observed above the lower grid. These observations by themselves are not conclusive except in identifying the weakest point in the incore string. If it is assumed that no other damage has occurred to the incore string during handling, and it is further assumed that TC junctions which reformed at a lower elevation did so because of being in contact with molten corium, a correlation between the instrument string break location and apparent new TC junction location would tend to confirm the analysis of TC junction data.

Video examinations of 51 incore instrument locations (Reference 23) after fuel assembly stub removal revealed that the incore string separated from the removed stub assembly at 37 locations and was observed to extend above the instrument guide tube in the lower grid. In 14 other locations, it was observed that the incore instrument string broke within the LCSA or the lower head area and was extracted with the stub assembly.

The break location of the instrument string is not a certain indicator of the status of the nozzle weld integrity. The instrument string break point could have resulted from other causes. For example, mechanical damage due to defueling operations or thermal distortion. However, it is important to note that there is a high degree of correlation between the observed break points and the TC data, as summarized in Table 2. This also tends to support the use of the TC data to conclude the nozzle welds are undamaged.

The video examinations are of special importance with respect to the eleven (11) locations, particularly incore nozzle location G-11, for which in situ tests showed the thermocouple to have an open junction (Reference Table 1), so that no thermocouple length reduction data exist. Since video data for all of these locations indicate that instrument string separation occurred within the core area, these data lend credence that incore damage does not extend to the lower head. Furthermore, the incore nozzles of the three (3) of the eleven (11) locations which have an open junction (i.e., H-13, F-13, L-13) have been visually observed (see Section 4.11.2); thus providing further credence that the nozzle welds at these locations have maintained their integrity.

4.11.5 Summary Information

The information presented in Sections 4.11.1 through 4.11.4 is summarized below:

a. Several conservative thermal analyses of the response of the RV lower head to the melted corium, both flowing and stationary, indicate little or no melting of the vessel wall would be expected.

b. Stress analysis of the vessel head indicates that failure due to creep would occur before temperatures sufficient to melt incore nozzle welds were achieved. Because no leakage from the vessel has been observed, vessel head creep did not occur and, therefore, the surface temperature of the vessel was well below melting.

c. Gamma scan data implies that a non-fueled layer of resolidified material may exist on the lower head inside surface; this would have protected the lower RV nozzle welds from melting.

d. Current visual examinations of incore nozzle welds at the outer periphery of the core have revealed no weld damage.

e. Incore instrument thermocouple resistance data indicate 41 measurable thermocouple junctions at or above the RV lower head. As indicated in Table 2, visual inspections during fuel assembly stub removal indicate a high degree of correlation between incore string length and thermocouple junction reformation measurements.

f. Visual inspections during stub assembly removal also indicate that the detectors for the 11 locations with open thermocouple junctions separated above the lower core support structure.

Other RV integrity safety concerns (e.g., assessment of potential damage to incore nozzles from pulling on incore instrument strings) are bounded by the evaluations provided by GPU Nuclear in Reference 19 as reviewed by the NRC in Reference 20.

Based on the above information, GPU Nuclear concludes that it is highly unlikely the incore nozzle to RV welds were significantly degraded during the accident. Thus, GPU Nuclear believes that LCSA/LH defueling can be conducted without impairing the integrity of the RV.

4.11.6 Precautions to be Exercised During LCSA/LH Defueling

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

4.12 Burning/Cutting Operations

Operation of burning devices inside the vessel has been evaluated by GPU Nuclear (References 2 and 6) and reviewed by the NRC (Reference 3). During initial LCSA dismantlement, 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 RV wall. As discussed in Reference 2, cutting operations are currently expected to begin on the top of the LCSA and sequentially cut through the lower grid rib assembly, lower grid flow distributor, lower grid forging, and incore guide support plate to the elliptical flow distributor. Thus, cutting of the elliptical flow distributor is precluded by the LCSA structure until the upper layers are removed. The elliptical flow distributor (which is more than one foot from the RV wall) will be cut only after considerable experience is gained by use of the plasma arc torch elsewhere in the RV. The arc or flame of such burning devices, operating underwater, will always be operated at least a foot from the RV wall. Because of rapid dissipation of the arc energy (i.e., within a few inches of water), propagation of an arc through one-foot of water is not possible. Thus, damage to the RV wall due to the operation of burning devices is precluded even when cutting the elliptical flow distributor.

4.13 Heavy Load Drops

During LCSA/LH defueling, the RV lower head and the incore nozzles will be subject to potential direct load drops. The RV lower head and incore nozzles will first be subject to a potential load drop during severing of the incore guide tubes. After the guide tube has been severed, it will be free to fall and impact the RV lower head debris bed, an intact incore nozzle, or the lower head proper. In addition, portions of the elliptical flow distributor may fall on the lower head.

After completing core bore operations, the incore guide tubes may be removed from the RV. This will create 52 approximate 6 1/2" diameter holes in the LCSA, each centered over an incore nozzle. These holes will provide a direct path for relatively small objects (i.e., long-handled tools) to impact an incore nozzle.

As mentioned previously, the remaining 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 RV. Eventually, following cutting of the elliptical flow distributor, a hole will be formed in the LCSA, exposing a large area of the RV lower head to a variety of different load drops.

Calculations have demonstrated that a load drop on an undamaged incore nozzle weld would not result in a nozzle weld failure (Reference Appendix A). The analysis in Section 4.11, of this report, demonstrates a high probability that the incore nozzle welds have maintained their original integrity. Additionally, there is a low probability that the configuration of a heavy load drop will directly impact an incore nozzle weld. Therefore, the potential for RV leakage due to dropped loads is remote.

There have been two instances of dropped loads involving the canister sleeve during defueling within the RV. Investigations by GPU Nuclear as to the cause of these drops have identified a failure of the sleeve locking drive retention spring which maintains the locking bar in position. In order to eliminate this potential failure mechanism, a new positive acting locking bar retention mechanism will be designed and installed before the elliptical flow distributor is cut.

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-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.

4.14 Reactor Building Basement

The potential for a criticality event in the Reactor Building basement was previously addressed in References 2 and 25.

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 LCSA/LH defueling. Therefore, criticality is precluded.

5.0 RADIOLOGICAL CONSIDERATIONS

Based on a comparison of activities associated with Reference 1 to those associated with LCSA/LH defueling, it is concluded that the radiological considerations associated with LCSA/LH 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 LCSA from the RV to their storage location within the Reactor Building. Although these pieces of the LCSA 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.

The sections of the LCSA to be removed under the scope of this SER are less radioactive than the lower grid rib assembly. The measured radiation level of a 5'x5' section of the lower grid rib assembly removed from the LCSA was 80 rem/hr within one (1) foot of the surface. At distance of 30 feet, the radiation level was less than 1 rem/hr following removal. This plate was rigged, moved, and unrigged remotely. Since the sections of the LCSA to be removed from the RV within the scope of this document will represent less of a radiation hazard, the adequacy of the personnel exposure control practices have been demonstrated by the lower grid rib assembly section removal.

Reference 1 estimated an occupational exposure to complete RV defueling of approximately 1400 person-rem. Currently, this estimate has not been exceeded. However, GPU Nuclear expects that the activities described in this SER and Reference 2 will cause this estimate to be exceeded. Thus, GPU Nuclear will provide an update to the expected occupational exposure to complete RV defueling and the jobhours and person-rem expended to date for defueling activities.

6.0 IMPACT ON PLANT ACTIVITIES

The major potential impact of LCSA/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 LCSA/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 b[illegible] 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 LCSA/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 LCSA/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. Reference 2 evaluates the consequences of potential events during LCSA/LH disassembly and defueling and demonstrates that LCSA/LH defueling can be performed safely.

This SER demonstrates that there is a high probability that the incore nozzles have maintained their original integrity; thus, the potential for a leak due to a load drop is not increased. Additionally, because a RV leak is not likely, the potential for fuel fines from the RV to migrate to the cavity beneath the RV in the Reactor Building basement due to an incore nozzle failure is remote. Further, Reference 2 demonstrates that a basement criticality event external to the vessel 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 References 1 and 2 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 LCSA/LH 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 LCSA/LH 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 Reference 2. 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 three (3) gallons of unborated water can be inadvertently drained into the RV (Reference 6).

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/LH defueling activities can be performed with no significant environmental impact.

9.0 CONCLUSIONS

Activities associated with LCSA/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. (Reference 1 provides the specific offsite dose analysis.) 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 LCSA/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. Safety Evaluation Report for Lower Core Support Assembly Defueling, Revision 2, 4710-3221-86-011, January 1988.

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

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

8. 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."

9. Hazards Analysis: Potential for Boron Dilution of Reactor Coolant System, Revision 0.

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

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

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

13. Evaluation of the Structural Integrity of the TMI-2 Reactor Vessel Lower Head - Final Report, June 1985, B&W 77-1154826-00.

14. EG&G-TMI-7784, August 1987, "TMI-2 Reactor Vessel Lower Head Heatup Calculations."

15. GPU TPO/TMI-175, Revision 6, "Analysis of Gamma Scanning of Incore Detector No. L-11 in Lower Reactor Vessel Head," June 1985.

16. EG&G-TMI-7811, September 1987, "Thermal Interaction of Core Melt Debris With The TMI-2 Baffle, Core Former and Lower Head Structures."

17. GPU Nuclear letter 4410-87-L-0189, dated December 28, 1987, "Use of Core Bore Machine for Dismantling the Lower Core Support Assembly," to Document Control Desk from F. R. Standerfer.

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

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

20. NRC Letter NRC/TMI-86-101, dated October 16, 1986, "Core Bore Operations," to F. R. Standerfer from W. D. Travers.

21. GPU Nuclear Technical Bulletin 88-02, Revision 0, dated April , 1988

22. GEND-INF-031 Volume II, April 1984, "TMI-2 Incore Instrument Damage - An Update."

23. GPU Nuclear Technical Bulletin 88-05, Revision 0, dated April , 1988.

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

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

TABLE 1

THERMOCOUPLE LENGTHS

Assembly Number	Grid Location	Original Length in Reactor (ft)	Calculated Reduction in Length (ft)	Length From Reactor Base (ft) *
1	H8	21.00	17.04	3.96
2	H9	20.97	6.91	14.06
3	G9	20.93	18.93	1.99
4	F8	20.86	19.70	1.16
5	E9	20.64	17.31	3.33
6	F7	20.82	19.58	1.24
7	E7	20.64	19.75	0.88
8	G6	20.82	20.04	0.78
9	G5	20.64	20.27	0.37
10	H5	20.68	16.65	4.02
11	K5	20.64	6.93	13.71
12	L6	20.71	8.95	11.76
13	M7	20.64	10.77	9.87
14	N8	20.41	9.84	10.57
15	N9	20.37	16.93	3.44
16	M9	20.64	19.97	0.66
17	M10	20.53	18.13	2.40
18	L11	20.53	7.96	12.56
19	K11	20.64	19.92	0.72
20	K12	20.37	--	--
21	H13	20.06	--	--
22	G13	20.02	13.49	6.53
23	F13	19.89	--	--
24	F12	20.26	6.63	13.63
25	G11	20.64	--	--
26	E11	20.33	20.33	0.00
27	D10	20.26	19.20	1.06
28	C10	19.89	10.77	9.12
29	C9	20.02	9.46	10.56
30	B8	19.59	9.52	10.08
31	B7	19.55	8.07	11.48
32	C6	19.89	8.42	11.48
33	D5	20.06	8.44	11.62
34	E4	20.06	--	--
35	F3	19.89	--	--
36	G2	19.55	8.92	10.[illegible]3
37	H1	19.00	7.30	11.70
38	L2	19.42	10.41	9.01
39	L3	19.89	10.48	9.42
40	M3	19.68	--	--
41	N4	19.77	7.81	11.96
42	O5	19.68	10.15	9.53
43	O6	19.89	--	--
44	P6	19.42	--	--

*These measurements have an uncertainty of ± 1.25 feet.
--Indicates open circuits.

TABLE 1

THERMOCOUPLE LENGTHS

Assembly Number	Grid Location	Original Length in Reactor (ft)	Calculated Reduction in Length (ft)	Length From Reactor Base (ft) *
45	R7	18.95	16.93	2.01
46	R10	18.80	6.58	12.22
47	O10	19.89	--	--
48	O12	19.37	8.41	10.96
49	M14	19.19	13.72	5.47
50	L13	19.89	--	--
51	D14	18.85	4.49	14.35
52	C13	18.95	8.37	10.58

*These measurements have an uncertainty of ± 1.25 feet.
--Indicates open circuits.

TABLE 2

CATEGORIZATION OF INCORE DETECTOR OBSERVATIONS

CATEGORY	NUMBER OF INCORES	CALCULATED LOCATION OF THERMOCOUPLE JUNCTION RELATIVE TO LOWER GRID RIB SECTION (a)	NUMBER & LOCATION OF DETECTOR SEPARATION BASED ON VIDEO DATA (ABOVE/BELOW LOWER GRID RIB SECTION)	AGREEMENT BETWEEN THERMOCOUPLE REDUCTION DATA AND VIDEO DATA %
A	23	ABOVE LOWER GRID	20-ABOVE LOWER GRID 2-BELOW LOWER GRID 1-UNKNOWN	87%
B	16	BELOW LOWER GRID	11-BELOW LOWER GRID 5-ABOVE LOWER GRID	69%
C	2	AT LOWER GRID(b)	2-ABOVE LOWER GRID	100%
-	*41*	*SUBTOTAL*	*33 OUT OF 41 AGREE*	80%
D	11	OPEN JUNCTION	11-ABOVE LOWER GRID	N/A
ALL	52	OVERALL	38-ABOVE LOWER GRID 13-BELOW LOWER GRID 1-UNKNOWN	80% (33 out of 41)

(a) BASED ON GEND-INF-031 Vol.II, April-84.
(b) MEASUREMENTS INCORPORATE AN UNCERTAINTY OF +/- 1.25 ft.

APPENDIX A

EVALUATIONS OF LOAD DROPS OVER THE REACTOR VESSEL

During core support assembly and lower head defueling, the lower core support assembly (LCSA) will have pieces cut from it and removed to gain access to core debris. Eventually, a hole will be created through the LCSA, exposing a large area of the RV lower 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 accidents 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"

* Distance to bottom, inside surface of RV lower head.
**Drops are sequential - first air then water.

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. In lieu of actually calculating drag coefficients for all dropped objects, a range of drag coefficients 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

*-Drag Coefficient

A comparison of the calculated allowable drop heights in Table B versus the maximum potential heights previously given in Table A shows that even for the very low drag coefficient (0.5) for objects A and B (i.e, the Light Duty Pole and the End Effector Handling Tool), the potential drop heights do not exceed the allowable drop heights. The loaded defueling canister with the minimum drag coefficient exceeds the allowable water drop height by about two (2) feet (34.1' vs. 36.6') and the loaded defueling canister with sleeve exceed the allowable height by about four (4) feet (19.6' vs. 24'). Note: Based on the maximum drag coefficient of 1.5, both objects have potential drop heights less than allowable drop heights.

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 enroute thereby decreasing its velocity. Further, the assumption that all of the impact energy will be 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 does 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. Unfortunately, in spite of this verification, canisters and sleeves have been dropped twice. Investigations as to the cause of these drops have indicated that the leaf spring which holds the locking device in place has failed. (Note: It is difficult to detect this hardware failure from the shielded work platform with the naked eye.) In order to preclude future drops to the fuel canister and sleeve combination, a new positive acting locking bar retention mechanism is being designed, tested, and installed prior to cutting the elliptical flow distributor. In the future, the position of this retention mechanism will be observed to insure that the locking device remains in the locked position. Consequently, the drop of a loaded canister and sleeve should have a low probability of occurrence.

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.