POLICY ISSUE
(Notation Vote)

October 9, 1990 SECY-90-350

For: The Commissioners

From: James M. Taylor
Executive Director for Operations

Subject: STATUS OF AND REMAINING ACTIONS FOR CLEANUP OF THREE MILE ISLAND NUCLEAR STATION, UNIT 2 (TMI-2)

Purpose: To provide the Commission with an update on the status of the TMI-2 cleanup and the remaining staff actions. Additionally, to request Commission approval of the staff's proposal to offer an opportunity for a prior hearing on a license amendment request that would allow for long term storage of the facility (Post-Defueling Monitored Storage or PDMS).

Discussion: Status

On January 30, 1990, almost 11 years after the TMI accident, GPU Nuclear Corporation (the licensee or GPUN) completed defueling of TMI-2. A detailed account of the current status of the cleanup is provided in Enclosure 1.

The licensee has determined that more than 99 percent of the core debris has been removed from the reactor vessel. The vessel has received a final flushing and brush-down, and the licensee has removed all remaining accessible fuel. On April 12, 1990, the licensee estimated that less than 850 kg of fuel remains at the facility. The licensee has completed an extensive video examination of the vessel and accessible areas of the reactor coolant system documenting the extent of fuel removal. Ex-vessel defueling has been completed. Decontamination efforts over the past year have been limited to maintaining the radiological conditions at the facility and minimizing occupational exposure.

Completion of Current Cleanup Effort

On February 22, 1990, the licensee submitted revision 5 to the Defueling Completion Report providing documentation of the completion of cleanup. The staff conducted a review to determine if the facility was sufficiently defueled. The staff evaluated the licensee's position that the facility was defueled to the extent that is reasonably achievable and that a criticality event is precluded.

Contact:
M. Masnik, PDNP, NRR
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NOTE: TO BE MADE PUBLICLY AVAILABLE WHEN THE FINAL SRM IS MADE AVAILABLE.

The results of the staff's review is contained in Enclosure 2. On April 26, 1990, the staff notified the licensee that it had satisfied the conditions for the cessation of defueling activities. This determination allowed the licensee to transition from Mode 1 through Mode 2 to Mode 3. Since the accident the staff and licensee have amended the TMI-2 technical specifications (TS) to reflect the safety significance of the facility as the cleanup progressed. Once the licensee demonstrated that defueling was completed and the possibility of a criticality was precluded the facility entered Mode 2 on April 26, 1990.

Upon transition to Mode 2 there was a substantial reduction in TS requirements, including the deletion of the requirement to man the control room with licensed operators.

Transition to Mode 3 occurred on April 27, 1990 after the staff verified that the licensee had shipped all fuel removed from the facility from the TMI-2 site.

Once defueling was complete, the licensee began the final phase of the current cleanup--readying the facility for long-term storage. Work scheduled for 1990 includes residual fuel measurements, preparation for and removal of water from the reactor coolant system and reactor vessel, and evaporation of accident generated water (AGW). Work will continue through 1991 on additional decontamination to meet target goals and removal of low-level wastes. The licensee plans to have the facility ready for PDMS by December 1992. A schedule for remaining activities is provided in Enclosure 3.

Post-Defueling Monitored Storage (PDMS)

On August 16, 1988, the licensee submitted a request for a license amendment that would allow for long-term storage of the facility (termed Post-Defueling Monitored Storage or PDMS by the licensee) once the plant reached a safe and stable condition. Although the licensee proposed that the amended facility license be a "possession only" license, the licensee has not expressly stated its intention to permanently cease operation or to decommission the plant. It is GPUN's position that, since it has not declared the permanent cessation of operations at TMI-2, the regulations pertaining to decommissioning are not applicable at this time. However, the licensee was required to submit a decommissioning funding plan as required by 50.75(b) and 50.33(k)(2).

The August 16, 1988 amendment request would further reduce the TS requirements consistent with the requirements for long-term storage. Enclosure 4 provides additional information on the licensee's proposal and the staff actions necessary to respond to the licensee's request.

The licensee plans to place the facility in PDMS until TMI-1 is ready for decommissioning. At that time, the licensee plans to decommission both Unit 1 and Unit 2 simultaneously.

In August 1989, the staff issued Final Supplement 3 to the Programmatic Environmental Impact Statement (PEIS) evaluating the impact of PDMS. The staff is currently preparing a safety evaluation report (SER) on PDMS. PEIS Supplement 3 and the staff's SER will form the basis of the staff's position on the acceptability of PDMS.

The staff proposes that an opportunity for a prior hearing should be offered on the licensee's PDMS license amendment request. The PDMS license amendment review is a detailed technical review principally involving the assessment of plant operating systems that will be necessary for the storage period. Much of the review is being performed under contract with Pacific Northwest Laboratories. The Commission held a prior hearing on the licensee's proposal to dispose of the accident generated water by evaporation. There was considerable public interest and involvement in the hearing. The licensee's current proposal to suspend further cleanup of the facility is considered by the public as a much bigger issue. Furthermore the Commonwealth of Pennsylvania is taking a much more active role in the PDMS review than it did in the accident generated water disposal issue.

Although no formal vote has been taken by the citizens Advisory Panel for the Decontamination of TMI-2, informal discussions with a number of members has revealed an expectation that a prior hearing would be held. If the staff announced that a hearing would not be offered prior to issuance of the PDMS amendment the Panel would likely vote overwhelmingly in favor of a prior hearing and insist that the Commission order a hearing prior to the staff's actions.

The licensee fully expects a prior hearing on this issue. The licensee has scheduled the remainder of the cleanup to coincide with a hypothetical schedule of a prior hearing. There would be little economic or schedular penalty to the licensee of a prior hearing versus a post issuance hearing. The TMI-2 Technical Specifications have been extensively modified since the accident to closely approximate those requested by the licensee for PDMS.

Based on the complexity of the technical issues associated with long term storage, the level of public and Commonwealth of Pennsylvania interest, and the lack of significant economic or schedular hardship on the licensee the staff recommends that an the opportunity for a prior hearing be offered on the PDMS license amendment request. If a hearing is required on this issue, it is estimated that the proceeding will take approximately 24 months. An Atomic Safety and Licensing Board Panel decision on PDMS would not be expected until late 1992.

The staff finds the licensee's application sufficiently complete at this time to go forward with issuance of the notice and the SER in the next couple of months.

Financial Information

As of the end of 1989, the licensee had spent $966 million on the cleanup. The licensee's budget for calendar year (CY) 1990 is $16.8 million. After the plant enters PDMS, the licensee expects to spend approximately $3 million to $5 million per year.

The licensee has stated that it will probably decommission TMI-2 along with TMI-1 when TMI-1's operating license expires. Under the current decommissioning rule, as long as the licensee maintains TMI-2 in a safe, stable configuration, additional cleanup before actual physical decommissioning would not be required. Therefore, additional cleanup at the conclusion of PDMS and before decommissioning would probably not occur unless it could be demonstrated that there is some health or safety reason requiring the additional decontamination that would offset the exposure necessary to accomplish the decontamination.

On July 26, 1990, the licensee submitted its decommissioning funding plan. According to this plan, the licensee will escrow $196 million for TMI-2 radiological decontamination. The licensee plans to have the full amount assembled at the time the TMI-1 license expires (April 19, 2014). The basis for the licensee's position is that it has not declared the permanent cessation of operations at TMI-2. The licensee asserts that such a declaration is a necessary prerequisite for the submittal of decommissioning plans and the need to fully fund its decommissioning fund account before the expiration date of its operating license (November 4, 2009). In its July 26, 1990 submittal, the licensee stated that it did not require an exemption to be allowed until the end of the TMI-1 operation to assemble the required funds because of the unique post-accident circumstances at TMI-2. Notwithstanding the licensee's proposal, the staff has developed an alternative position on the issue of when funds must be available for decommissioning for facilities that prematurely cease operation. This position will be forwarded to the Commission in a forthcoming Commission Paper.

TMI-2 Advisory Panel

The Advisory Panel for the Decontamination of TMI-2 last met on March 14, 1990. The panel discussed several topics including whether it is necessary to continue meeting now that defueling is completed. Several panel members expressed interest in continuing to meet, although less frequently and perhaps with fewer panel members.

At the March 14, 1990 meeting, the panel decided to meet again in the fall and continue discussions on the continuance of the panel. Other topics planned for discussion at the proposed October 1990 meeting are disposal of the AGW (evaporator operating experience), the licensee's decommissioning funding plan, the staff's PDMS SER, and the future role of the Advisory Panel.

The staff recommends that the Advisory Panel meet with the Commission sometime this fall to discuss the panel's activities since the last meeting with the Commission on October 25, 1988, and to discuss the future of the panel.

Disposal Of Accident-Generated Water

On September 11, 1989, the staff granted an amendment to the TMI-2 license removing the prohibition to evaporate the accident-generated water (AGW). This amendment was issued after a lengthy hearing proceeding. On January 19, 1990, the Atomic Safety and Licensing Appeal Board affirmed the Atomic Safety and Licensing Board's decision. The licensee has installed the evaporator on site and has begun surrogate testing with water containing boron and sodium to test the complete performance of the system from the evaporation process through the processing of the concentrate and solids. There have been a number of equipment breakdowns during testing, resulting in repairs and several minor design modifications. These breakdowns have delayed surrogate testing and caused significant delays in readying the evaporator for operation. Intervenors have also challenged the Commonwealth of Pennsylvania and the licensee regarding the issuance of an air quality waiver for evaporation of the AGW.

Fuel Measurements and Special Nuclear Material (SNM) Accountability

The staff has contracted with Pacific Northwest Laboratory to conduct independent measurements to determine residual fuel remaining at the TMI-2 facility sometime during the summer of 1990. Preliminary measurements of the Auxiliary and Fuel Handling Building were taken in May 1990. The results of the measurements program will be needed to support the staff's testimony at a likely hearing on PDMS and will also form the basis of a policy statement by the Commission on the conclusion of the current TMI-2 cleanup effort.

The staff, with contractor support, is also conducting a review of the basis for the special nuclear material transfer documents transferring accountability for the fuel to the U.S. Department of Energy (DOE).

Lower Head Sampling

The Office of Research's contractor, MPR Associates, Inc., completed a sampling program of the reactor vessel lower head (Enclosure 5) earlier this year. Prism-shaped metallurgical samples were removed for analysis.

Conclusions: Defueling of TMI-2 has taken over 4 years and has not proceeded as quickly as originally anticipated. Consistent with past Commission direction, the staff has reviewed the reasons for defueling delays and concludes that the delays are based on technical considerations associated with this unique activity and are not a result of either a lack of management commitment or lack of available funds. Additionally, the cleanup has not been delayed as a result of NRC staff regulatory activities. Consistent with the Commission's policy of supporting an expeditious cleanup, the Agency's licensing and inspection staff has continued to manage its oversight activities within the licensee's schedules. The staff has also continued to keep the Commission's Advisory Panel and the public informed of the progress of the cleanup.

Although the staff believes that some additional delay in placing the facility in long-term storage (PDMS) is likely, the delay will have little safety significance. Once the staff determines

that long-term storage of the facility will not jeopardize public health and safety, then the staff is justified in concluding that the current phase of the cleanup has been accomplished.

Consistent with past practice, the staff will keep the Commission informed of the progress of the cleanup.

Recommendation: The Commission approve the staff's recommendation that a notice for an opportunity for a prior hearing on the PDMS license amendment request be issued by the staff.

James H. Snaezek /for
James M. Taylor
Executive Director
for Operations

Enclosures:
1. Status of the Cleanup of TMI-2
2. Staff Review of the Completion of Defueling
3. Cleanup Schedule for TMI-2
4. Post-Defueling Monitored Storage of TMI-2
5. Lower Head Sampling Program

Commissioners' comments or consent should be provided directly to the Office of the Secretary by COB Wednesday, October 24, 1990.

Commission Staff Office comments, if any, should be submitted to the Commissioners NLT Wednesday, October 17, 1990, with an information copy to the Office of the Secretary. If the paper is of such a nature that it requires additional time for analytical review and comment, the Commissioners and the Secretariat should be apprised of when comments may be expected.

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Enclosure 1

Status of the Cleanup of Three Mile Island Unit 2

Plant stabilization of and cleanup efforts on the Three Mile Island Nuclear Station, Unit 2 (TMI-2), began immediately after the accident March 29, 1979. In the succeeding 11 years, the GPU Nuclear Corporation (the licensee) has made substantial progress in placing the facility in a safe, stable condition that poses little or no threat to the public health and safety. The following discussion on the status of the cleanup will describe the defueling of the reactor vessel; the defueling of the remainder or the primary system (ex-vessel defueling); the decontamination of the reactor and auxiliary and fuel-handling buildings; the fuel cask shipments; and the disposal of the accident-generated water (AGW). During these activities, approximately 99 percent of the original core mass was removed.

Defueling of the Reactor Vessel

Defueling of the reactor vessel is essentially complete. On December 16, 1989, following completion of pick-and-place defueling and vacuum defueling of the lower head, the licensee declared the end of bulk defueling. All areas of the reactor vessel had been defueled at least once. Some of the defueling operations (the pick-and-place defueling and air lifting) redistributed fines on horizontal surfaces and small debris into crevice-shaped areas.

After completing bulk defueling, the licensee began a "flush-and-brush" process and final vacuuming to remove this redistributed material. On January 30, 1990, the licensee completed the flush-and-brush process, final vacuuming, and a video confirmation of the defueling.

During February 1990, the metallurgical sampling program was completed. Following this activity, the defueling crews performed a final series of pick-and-place defueling, vacuuming and confirmatory video examinations. On March 20, 1990, the crews transferred the last canisters of core debris to the fuel handling building.

During the cleanup, these crews removed a total of approximately 308,000 pounds of core debris from TMI-2. The 308,000 pound mass included the mass of the core, structural, and absorber materials; the mass added by the oxidation of the core and structural materials; and the portions of baffle plates, formers, and other components that had commingled with core debris during defueling operations.

The total quantity of residual fuel (UO_2) estimated remaining in the reactor vessel after the conclusion of defueling is 609 Kg.

Ex-vessel Defueling

Cleanup crews have completed fuel removal from the ex-vessel portions of the reactor coolant system (RCS). These areas include the once-through steam generators (OTSGs), the pressurizer, the RCS hot legs, the decay heat drop line, and the core flood penetrations.

The total quantity of residual fuel (UO_2) estimated remaining after the conclusion of defueling in the reactor coolant system (excluding the reactor vessel) is less than 133kg; the residual fuel in the reactor building (excluding the reactor coolant system) is less than 75kg; and the residual fuel in the auxiliary and fuel handling building is less than 17kg.

Fuel Cask Shipments

Core debris were removed from the reactor vessel in canisters, placed into fuel shipping casks, and shipped by rail to the Idaho National Engineering Laboratory. The final shipment was made on April 15, 1990.

Building Decontamination

From early December 1988 until the end of defueling, the licensee focused its efforts on the completion and support of defueling. Decontamination activities were limited to support defueling, and to maintain access to and operability of plant systems. Decontamination and building cleanup activities required to place the facility in post-defueling monitored storage have resumed following the completion of defueling.

Disposal of Accident-Generated Water

On September 11, 1989, the staff of the U.S. Nuclear Regulatory Commission (NRC) issued a license amendment removing the prohibition on disposal of the AGW by use of an evaporator system. The staff issued this license amendment after a 2-year review of the licensee's proposal that included a public hearing before an NRC Atomic Safety and Licensing Board Panel. The licensee has been testing the evaporator system with non-radioactive surrogate solutions. The testing has proceeded slowly because of equipment failures and the time spent by the licensee on equipment upgrades and modifications to improve the long-term reliability of the system. The licensee has completed testing and will likely initiate evaporation of the AGW in September 1990.

Enclosure 2

Staff Review of the Completion of Defueling

Since the accident, the staff of the U.S. Nuclear Regulatory Commission (NRC) and GPU Nuclear Corporation (GPUN; the licensee) have amended the Technical Specifications to reflect the continuing decrease in safety significance of the facility as the cleanup progressed. On May 27, 1988, the staff issued License Amendment 30 that allowed the licensee to transition through a series of modes as specific milestones in the cleanup were achieved. As the cleanup progressed, the licensee could transition to the next, less restrictive mode. With the transition to each mode, this amendment would reduce the technical specification requirements commensurate with the increase in safety. When Amendment 30 was first issued, the licensee was to begin in Mode 1, and end with Mode 3 after completing defueling and shipping all the fuel. Transitioning to Mode 2 and Mode 3 was on specific cleanup criteria specified in the Technical Specifications. The licensee was required to document their defueling performance and provide the justification in the Defueling Completion Report that the specific criteria were met and that they could progress to Mode 2. The staff had 60 days to review the licensee's submittal before the licensee passed to the next mode. Section 1.3 and Table 1.1 of the Technical Specifications require the following three conditions for the licensee to move from Mode 1 to Mode 2:

a. The Reactor Vessel and Reactor Coolant System are defueled to the extent reasonably achievable.

b. The possibility of criticality in the Reactor Building is precluded.

c. There are no canisters containing core material in the Reactor Building.

On February 22, 1990, the licensee submitted the Defueling Completion Report, Revision 5, documenting the completion of the defueling effort. This report was supplemented on April 12, 1990.

In a memorandum to J. Stolz of April 26, 1990 (Attachment 1), the staff concluded that the facility has been defueled to the extent reasonably achievable. Under contract with NRR, Pacific Northwest Laboratories concluded in a letter to M. Masnik, NRC, on April 13, 1990 (Attachment 2) that there is no potential for criticality in the fuel remaining in the TMI-2 facility during either normal or accident conditions. On April 26, 1990, the staff issued a memorandum to J. Stolz, NRC, (Attachment 3) verifying that no canisters containing core material remained in the reactor building, thereby satisfying all three conditions for the transition from Mode 1 to Mode 2.

The April 26, 1990, memorandum (Attachment 3) from the staff to J. Stolz, also verified that no canisters containing core material are stored at the TMI-2 site. This fulfilled the requirement for the transition from Mode 2 to Mode 3.

On April 26, 1990, the staff issued a letter to M. Roche, GPUN, stating that the staff had no objection to the licensee transitioning from Mode 1 to Mode 2 on April 26, 1990, and from Mode 2 to Mode 3 on April 27, 1990.

Attachments:
1. Memo to Stolz 4/26/90
2. Ltr to M. Masnik 4/13/90
3. Memo to Stolz 4/26/90

Attachment 1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

April 26, 1990

Docket No. 50-320

MEMORANDUM FOR: John F. Stolz, Director
Project Directorate I-4
Division of Reactor Projects - I/II

FROM: Michael T. Masnik, Senior Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II

Lee H. Thonus, Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II

Francis I. Young
Senior Resident Inspector
Region I

SUBJECT: REVIEW OF LICENSEE'S FINAL RESIDUAL FUEL ESTIMATES
THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 2 (TAC 74532)

On March 13, 1990 and again during April 2 through April 5, 1990, the staff conducted a review of GPU Nuclear Corporation (the licensee) video inspection tapes and documents supporting the licensee's estimates of residual fuel at the TMI-2 facility.

The purpose of the review was twofold: 1) independently verify, on an audit basis, the licensee's estimates of fuel remaining at TMI-2 at the conclusion of the current cleanup effort, and 2) determine if the licensee removed as much fuel as is reasonably achievable. This effort is part of the review of the licensee's proposal to transition the TMI-2 facility to Mode 2 as detailed in their Technical Specifications.

Section 1.3 and Table 1.1 of the Technical Specification require the following three conditions for the licensee to transition from Mode 1 to Mode 2:

a) The Reactor Vessel and Reactor Coolant System are defueled to the extent reasonably achievable.

b) The possibility of criticality in the Reactor Building is precluded.

c) There are no canisters containing core material in the Reactor Building.

The licensee submitted the Defueling Completion Report (DCR), Final Submittal on February 20, 1990. Additionally, the licensee submitted on April 12, 1990, the results of their post-lower head sampling program cleanup. These two documents form the basis of the licensee's proposal for transitioning to Mode 2.

This memorandum provides the staff's review of the licensee's submittal addressing conditions a and part of b above. This memorandum will address

condition b first. In order to evaluate the potential for a criticality the staff had to verify the location, amount, and condition of the remaining fuel at TMI-2. Since the fuel is located in a number of locations in the Reactor Building (RB) and the Auxiliary and Fuel Handling Building (AFHB), the staff determined that an audit of the remaining fuel was warranted.

The staff first evaluated the possible locations for significant quantities of residual fuel. The staff determined based on an understanding of the accident progression, 10 years of defueling oversight, and the licensee submittals that significant remaining quantities of fuel would be confined to the RB and principally to the reactor coolant system (RCS). Little fuel was dispersed to the AFHB and the RB outside of the RCS during the accident and subsequent cleanup.

The staff then examined the potential for the licensee to have overlooked significant quantities of fuel. The staff reviewed schematics of the RCS and in particular the reactor vessel (RV) for locations where fuel that could be present might have been overlooked by the licensee. Since the RV was physically the most complicated structure in the RCS and the balance of the RCS had been well characterized or measured, the staff concentrated on the RV. A total of four locations within the RV with the potential for uninventoried fuel were selected and video tapes of the locations were inspected:

a) Top edge of the thermal shield between support blocks. The licensee had not included the fuel located on the top edge of the thermal shield in between the support blocks in their estimate of fuel remaining at TMI-2. The staff estimated the fuel remaining at 0.6 kg with an upper limit of 1.2 kg.

b) Inner surface of the core barrel. The inner surface of the core barrel was essentially free of all Type I or Type III fuel (Type I is loose debris, Type III is solidified material). One small area, approximately 1" x 2" x ½", of possible Type III material was observed near core former plate Level 8. This represents approximately 84.9 g of fuel.

c) Top edge of the vent valves on the inside and outside of the core support shield. Top edge of the vent valves outside the core support shield had been examined by the licensee and the remaining fuel included in the total for the core support structure. The top edge of the vent valves and the top edge of the vent valve horizontal restraining blocks were not included in any estimate provided by the licensee; however, a review of several photographs and at least one video tape indicated that the slope of the surfaces would not permit the buildup of more than gram quantities of fuel in these areas.

d) Edge holes of the former plates. The edge holes of the former plates that formerly contained the baffle plate retaining bolts were examined and no residual fuel was observed in any hole.

The staff determined that there was some additional fuel that the licensee failed to include in their estimates; however, the amounts were insignificant (probably less than 1 kg) relative to the reported quantities (608.8 kg) of fuel in the RV. The staff decided to enlarge the audit sample size and examine additional areas; however, based on additional review of the RV schematics, no other locations that could possibly hold significant (greater than 4 kg) of fuel were identified. Therefore, the staff concluded that the licensee had, with the exceptions noted above, looked and attempted to characterize remaining fuel in all locations within the RV.

The ex-vessel regions of the RCS have been characterized using a combination of video inspections and actual fuel measurements. The staff concluded that the licensee identified all the areas which contained or could contain significant quantities of residual fuel.

The staff then reviewed the licensee's characterization and quantification of residual fuel in several designated areas. A total of five areas were selected, four in the RV.

a) Lower Head region of the reactor vessel. Since it is the lowest portion of the reactor vessel, finely divided loose debris put into suspension by water currents tends to preferentially settle out on the lower head. Loose, sand-sized particles generated from defueling activities also tend to drift downward and settle on the lower head. The staff reviewed the final video of the lower head and the licensee's analysis of the remaining fuel on the lower head. The licensee's assumptions in their engineering calculation 4800-3221-90-013 were more conservative than "best estimate" and the calculations were numerically correct. The staff performed independent calculations which conservatively modelled the remaining fuel debris dust as a series of three ellipsoid areas. The licensee had modelled a thin coating of material which covered the entire lower head surface. The staff's results were 7.4 kg of fuel and the licensee's results were 8.1 kg of fuel.

b) Annular gap region between the core barrel and the thermal shield. The licensee reported 118.6 kg of fuel in this area. It represents the largest amount of Type I (loose, finely divided) fuel remaining at TMI-2. The area is very difficult to access; a miniature (.75 inch) camera was successfully inserted in 3 of the 4 locations where vent valve exercise holes aligned with gaps in the thermal shield restraint blocks. Finely divided fuel debris filled the bottom of the annular gap. This bottom region has a complex, wedge shaped cross section. The staff reviewed the licensee's assumptions; they conservatively used the maximum depth observed by the 3 camera probes. The staff also independently verified the dimensions of the area from detailed plant drawings and checked the accuracy of the licensee's calculations.

c) Lower Core Support Assembly (LCSA) between the Lower Grid Forging and the Incore Guide Tube Support Plate (IGTSP). This area contains the largest quantity (133 kg) of residual fuel in any one remaining area at TMI-2. It includes a single resolidified fuel mass (Type III) of 110 kg. It is the dominant component in the criticality model. The staff reviewed video tapes of this area and verified that it was accurately to conservatively characterized. There were some irregular interstitial gaps in the mass which the licensee conservatively characterized as a monolith.

d) Inside and outside surfaces of the core baffle plates. During the accident progression in 1979, a molten mass from the core region melted a large irregular hole (approximately 5 ft by 2.5 ft) through the baffle plates in the southeast region of the core. The molten material spread radially behind the core baffle plates and downward to the LCSA and the lower head. Some of the molten material stuck to the baffle plates as it cooled and resolidified. The licensee analyzed the amount of fuel remaining on the inside (closest to the core) and assumed that an equal amount adhered to the outside (closest to the baffle plates) surface. The staff was concerned about this assumption. The licensee's calculations and assumption were based on a September 1989 video examination. This video was of the inside surface, thus the assumption. A subsequent video inspection taken In October 1989 (apparently after additional cleaning) showed both sides of the baffle plates and demonstrated that the licensee's assumption was conservative.

e) Pressurizer spray line. The licensee reported that there was no fuel in the pressurizer spray line because the line had been flushed. The staff was concerned about the validity of this assumption. Further investigation revealed that the licensee had evaluated the spray line relative to the pressurizer surge line, which had been measured for fuel content. The surge line is much larger (10 inch diameter vs. 4 inch), contains a longer run of horizontal piping, and was much more directly involved in fuel transport during the accident. It was unflushed and measurements indicated only 0.2 kg of fuel. The staff concluded that the spray line would contain less than 0.05 kg of fuel and that this amount of fuel could be considered negligible compared to the 608.8 kg remaining at TMI-2.

The staff has determined after reviewing the video tapes of selected locations within the reactor vessel and the calculation of residual fuel based on the video inspections that the estimates of residual fuel presented in the DCR as updated in their April 12, 1990 submittal are reasonable and conservative.

The staff has also reviewed the licensee's procedures for quantifying the fuel and find the licensee's approach to quantifying the fuel acceptable and conservative. The staff discussed and reviewed correspondence between TMI-2 licensing and the TMI-2 Safety Review Group (SRG) regarding the SRG review of the fuel measurements. The staff has concluded that the SRG provided an appropriate and comprehensive review of the licensee's fuel estimation program. SRG concluded that the DCR reported fuel quantities and the subsequent re-evaluations of residual fuel conservatively estimate the quantity of fuel in the RV.

During the course of the staff's review of the licensee's estimates of residual fuel the staff also evaluated the first condition of the three conditions necessary to transition to Mode 2. The first condition states that "the reactor vessel and reactor coolant system are defueled to the extent reasonably achievable".

The staff has not identified any technical limitations that would preclude additional fuel removal; however, the removal of additional fuel would exact a progressively greater penalty from the standpoint of occupational exposure and cost as more difficult areas are defueled. The staff has determined, based on the review of the final video inspections of the RCS that the fuel that could be easily removed using the present defueling technology has been removed. Admittedly, there is some additional Type I loose debris on the bottom head; however, this represented less than 11.2 kg of debris and resuspension of the material during removal would make removal of all the material on the bottom head almost impossible. The remaining areas of the residual fuel and in particular the significant amounts of residual fuel have been determined to reside in areas that are difficult to access. The difficulty in access limits visibility and the manipulation of defueling equipment. The licensee provided an analysis in the DCR which addressed each significant quantity of fuel within the facility. Significant quantity of fuel was defined by the licensee as greater than 10% of the licensee's calculated safe mass fuel limit (SMFL). Since the licensee's SMFL is 140 kg the licensee's definition of significant quantity of fuel is 14 kg. The licensee identified the following areas as having significant quantities of fuel: RB "A" D-ring, "B" OTSG tubesheet, cold leg 2A, and the RV.

The licensee then discussed each significant quantity of fuel addressing the quantity, the potential options for additional fuel removal, schedule impacts for fuel removal, occupational exposure, waste generation and disposal, and cost. The licensee's analysis concluded that the TMI-2 facility had been defueled to the extent reasonably achievable.

The staff has reviewed the licensee's submittal, the final video inspection of the facility, and the measurements of residual fuel made by the licensee, and have determined that:

- The licensee has defueled to a point that additional defueling will become progressively more difficult due to accessibility and reduced quantities of fuel.

- Additional defueling will result in increased person-rem exposure and cost per kg of fuel removed and possibly increasing risk to cleanup personnel without a measurable reduction in risk to the public. The risk to workers include continued occupational radiation exposure, the potential for overexposure, industrial accidents and continued physical stress due to the hostile working environment.

With the possibility of a criticality precluded, and the elimination of any energy source sufficient to cause an offsite release of radioactivity in excess of the Appendix I guidelines, the staff has concluded that continued defueling at this time would not be justified based on ALARA consideration and overall risk to the workforce. Therefore, the staff has concluded that the facility has been defueled to the extent reasonably achievable.

This review was conducted by M. Masnik, NRR, L. Thonus, NRR, and F. Young, RI.

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April 13, 1990

Dr. Michael T. Masnik
Office of Nuclear Reactor Regulation
MS 13 H3
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Dr. Masnik:

A report of PNL's review of the TMI-2 licensee's Defueling Completion Report is enclosed. PNL's report incorporates information received in the licensee's letter to the NRC dated April 12, 1990 (Subject: Results of Post-Lower Head Sampling Program Cleanup), however, as we have discussed on the telephone, PNL will continue reviewing the licensee's letter and will notify you of the results of our continued review by early next week.

Sincerely,

Rocky Harty

R. Harty
Senior Research Scientist
Dosimetry Research Section
HEALTH PHYSICS DEPARTMENT

RH/lms

Enclosure

cc: KS West, US NRC, w/enclosure

REVIEW OF THE DEFUELING COMPLETION REPORT FOR TMI-2

1.0 INTRODUCTION

As a result of the March 28, 1979 accident at Three Mile Island Nuclear Station, Unit 2, fuel debris was transported and relocated within the reactor coolant system, the reactor building and the auxiliary and fuel handling buildings (AFHB). Many of the cleanup activities have concentrated on the removal of the remaining fuel debris from within the reactor vessel as well as that transported ex-vessel.

On May 27, 1988, the NRC issued License Amendment No. 30, which defined three facility modes for the TMI-2 facility. This amendment established that 60 days prior to transition to each successive facility mode, a report shall be submitted to the NRC providing the necessary basis and justification for the transition. For transition from Mode 1 to Mode 2, the licensee is required to demonstrate that;

1. the reactor vessel and reactor coolant system (RCS) have been defueled to the extent reasonably achievable,

2. the possibility of criticality in the reactor building is precluded, and

3. that there are no canisters containing core material remaining in the reactor building.

In conjunction with the issuance of License Amendment No. 30, the NRC granted GPU Nuclear an exemption from 10 CFR 70.24 for the criticality monitoring requirements in the TMI-2 reactor building. This action imposed the following mode transition provision:

> "Prior to transition to Mode 2, the licensee will provide a criticality analysis that will address each separate quantity of residual fuel in each defined location. The criticality analysis will estimate the quantity of fuel remaining, its location, its dispersion within the location, its physical form (i.e., film, finely fragmented, intact fuel pellets), its mobility, the presence of any mechanism that would contribute to the mobility of the material, the presence of any moderating or reflecting material, and its potential for a critical event. In this submittal the licensee must demonstrate that the cleanup has progressed far enough that an inadvertent criticality is precluded."

By letter dated February 22, the licensee submitted to the NRC the fifth and final submittal of the Defueling Completion Report. The report was supplemented by a letter dated April 12, 1990 containing the results of the cleanup following the lower head sample program and containing a revised criticality analysis that made use of the January 1990 video inspection results. The report as supplemented provides the licensee's estimate of the quantity of fuel remaining, its location, its form, any potential for mobility, and the potential for criticality. The NRC has requested that PNL

review the criticality aspects of the Defueling Completion Report and the measurement methods that were used. This report is a summary of PNL's review.

2.0 DESCRIPTION OF THE LICENSEE'S DEFUELING COMPLETION REPORT

The licensee's Defueling Completion Report contains the licensing background behind the development of the report, a description of the post-accident fuel dispersion process, the techniques that were used to survey for and measure the quantities of fuel remaining in the facility, the procedures and methods that were used to remove fuel from its post-accident locations, defueling occupational dose estimates to date and estimates of the occupational dose that would be required for continued removal of fuel. The Defueling Completion Report provides a quantification of the amount of fuel in all areas of the facility, and provides an assessment of the criticality potential in each of the areas. The facility is divided into four major areas; 1) the auxiliary and fuel handling building (AFHB), 2) the reactor building (outside of the reactor coolant system and the reactor vessel), 3) the reactor coolant system (RCS), and 4) the reactor vessel. Individual locations within these larger areas were evaluated in detail.

The evaluation of the potential for a criticality in the remaining fuel debris was made by determining the safe fuel mass limit (SFML) for the TMI-2 fuel, that is, the quantity of fuel below which there would be no possibility of an accidental criticality. Calculation of the SFML assumed an optimum fuel geometry (size and shape of the fuel), optimum moderation and an infinite water reflector. This provides the optimum conditions for a criticality. For locations where the quantity of fuel exceeded the SFML, a more realistic model of the fuel and of the fuel's location was created for analysis. The only location where a specific model was necessary was in the reactor vessel.

3.0 EVALUATION

The Defueling Completion Report was evaluated by reviewing the report as submitted by the licensee as well as by reviewing information referenced within the Defueling Completion Report. PNL's review primarily encompassed the measurement techniques used to estimate the quantity of fuel and the criticality model used to determine whether an inadvertent criticality could occur.

3.1 ANALYSIS OF FUEL MEASUREMENTS

The measurements techniques used to estimate the quantity of fuel remaining following defueling were analyzed by PNL. A review of those estimates resulting from visual examination of the remaining debris was conducted by the NRC staff. Questions involving specific measurement techniques were resolved verbally with GPUN engineers and contractors. PNL found no problems with the measurement techniques used by the licensee.

3.2 SAFE FUEL MASS LIMIT (SFML)

The licensee evaluated the potential for an inadvertent criticality in the remaining fuel by determining the SFML for the TMI-2 fuel. The SFML was calculated using the assumptions discussed below. The SFML was then applied to the locations where fuel remained to ensure that the fuel in these locations would not go critical even under optimum conditions.

3.2.1 Assumptions used in the Determination of the SFML

The SFML was calculated to be 140 kg. This quantity was used to compare with the quantity of fuel estimated for each location where fuel remained. The assumptions used to calculate the SFML are given below, along with an analysis of their appropriateness:

3.2.1.1 The fuel enrichment was assumed to be 2.54% before burnup.

The original loading of the core included 56 assemblies of 1.98 wt% (batch 1), 61 assemblies of 2.64 wt% (batch 2), and 60 assemblies of 2.96 wt% (batch 3) uranium-235 enrichment. The licensee assumed a homogeneous mixture of the three fuel batches based on samples obtained following the accident that indicated that most of the residual fuel in the TMI-2 reactor vessel and ex-reactor components is a homogeneous mixture of the pre-accident core loading. In addition, the licensee indicated that approximately 65% of the batch 3 fuel was removed from the vessel as intact full or partial length fuel assemblies without significant mixing. Thus a pre-burnup enrichment of 2.54 wt% uranium-235 was used by the licensee to calculate the SFML. Although the assumption of a homogeneous mixture is well made, especially in light of the additional redistribution and mixing of the fuel that occurred during defueling activities, the sampling program in the reactor vessel indicated a wide variation in enrichment data even among samples collected in close proximity to one another (as shown in Tables 3 and 4 of Appendix B to the Defueling Completion Report). Thus, it is possible that regions containing significant quantities of material from the more reactive, higher enrichment of 2.96 wt% uranium-235 may exist. The assumption that the remaining fuel has an enrichment of 2.96 wt% uranium-235 prior to burnup would have assured that any such region is accounted for in the criticality analysis. Although this assumption would be highly conservative, the possibility of localized areas with higher enrichments can not be dismissed.

3.2.1.2 Fuel burnup effects were considered for all three batches of fuel.

The incorporation of the fuel burnup effects are a realistic assumption and one that is fairly well

calculated. The effect of the burnup was to reduce the fuel enrichment used by the licensee to 2.24 wt% uranium-235 (or 2.67 wt% uranium-235 assuming a batch 3 enrichment). Although no attempt was made by PNL to reproduce the burnup composition used for the criticality analyses, the ORIGEN code that was used by the licensee to determine post-accident fission product concentrations, uranium depletion and plutonium buildup is an accepted and widely used computer code for this purpose. Considering that fuel burnup is not uniform over the core or from rod to rod, the fuel composition used in the criticality analyses are in good agreement with extrapolated compositions similarly calculated for the H. B. Robinson Unit 2 reactor fuel cycles 1 and 2 (Barner, 1985), which have an enrichment/loading pattern very similar to TMI-2.

The approach used by the licensee to account for the effects of burnup are conservative in that the analysis relies on only a limited few of the non-volatile, insoluble fission products that are relatively strong neutron absorbers (neutron poisons). For example, europium-155 decays to gadolinium-155 which becomes a significant source of neutron absorption after several years. However, gadolinium and other neutron absorbing fission products were not relied upon in the licensee's analysis to reduce the reactivity of any of the regions containing fuel. The licensee only accounted for the poisoning effects of fission products that were definitively identified as remaining with the fuel.

3.2.1.3 The equivalent of standard, full sized fuel pellets were used for the fuel particle size.

The residual fuel debris in the TMI-2 facility exists in three forms: 1) resolidified material, 2) thin surface films and 3) loose granular debris. The resolidified material is too large for optimum neutron moderation and the surface films are too thin and distributed to be of concern unless they flake off and become a part of the loose granular debris. Since optimum moderation is dependent on the particle size of this debris, the licensee performed calculations to demonstrate that particles smaller than a fuel pellet resulted in less than optimum conditions for criticality. The results are in agreement with data given by Clark (1967) in DP-1014. Consequently, the mass-to-surface ratio of a normal fuel pellet was preserved in the geometrical modelling of the debris. Conservatively, it was assumed that nothing was present in the fuel except spherical pellets of fissile material and unborated water. These conditions are optimum for the criticality of debris having mass-to-

surface ratios up to that of a TMI-2 fuel pellet, and the fuel pellet diameter is essentially optimum.

3.2.1.4 No credit was taken for the presence of structural and solid poison materials existing in the fuel debris.

Impurities that have been identified as existing with the fuel (for example boron, iron, zirconium, and cadmium) were not included in the analysis of the SFML. As mentioned earlier, the only poisons assumed for the analysis are those associated with the burnup of UO_2 and include only the effects of fissionable plutonium generation and rare earth fission product generation.

3.2.1.5 Maximum reflection and moderation were considered.

A spherical geometry was chosen to maximize the neutron multiplication and an effectively infinite water reflector was utilized. The model used to calculate the safe fuel mass consisted of a mixture of unborated water and fuel debris in a spherical geometry surrounded by an infinite thickness (approximately 12 inches) of unborated water reflector. The radius of the innermost region was varied until the calculated k_{eff} (including a 2.5% Δk uncertainty bias - as discussed below) reached 0.99. An acceptance criteria for k_{eff} of 0.99 was used during the post-accident defueling operations at TMI-2. Although an acceptance criterion of 0.95 (based on the limit allowed in standard technical specifications for spent fuel storage) is applicable to conditions during Mode 2, the licensee's calculation of maximum reflection and moderation is still highly conservative and optimizes the calculated potential for criticality.

3.2.1.6 The computer code KENO V.a was used and an uncertainty of 2.5% Δk was added to the calculated k_{eff}.

The criticality calculations for the SFML were performed using the KENO V.a criticality code and the 27 group ENDF/B-IV cross section library of the SCALE system. These are the accepted computer codes for criticality analyses. The bias of 2.5% in k_{eff} added to all calculated k_{eff} values obtained with the KENO/SCALE systems is based on the ability of this calculational technique to reproduce twelve criticality experiments chosen for this purpose. Although these assemblies were selected to provide benchmark data on conditions similar to those resulting from the high boron-low hydrogen content of the TMI-2 fuel rubble, a more representative group of experiments should, and could, have been selected. Some of the large disagreements between the calculations and the experiments selected is likely due to

these poor choices. However, except for highly borated systems, the 2.5% bias used in the criticality analyses is approximately the 2 sigma bias observed in a great many comparisons over the years between experiments and KENO/SCALE calculations. The technique tends to calculate boron systems about 2% low in k_{eff}. Thus, the 2.5% bias is acceptable, especially since the criticality analyses assumes that there is no boron in the fuel rubble.

3.2.2 Calculation of the SFML

The SFML reported in the Defueling Completion Report was determined by calculating the allowable mass that was approximately 75% of the minimum calculated critical mass (k_{eff} = 1.0). The SFML was calculated as 140 kg UO_2. Utilizing data published by H. K. Clark (1967) in DP-1014, "Critical and Safe Masses and Dimensions of Lattices of U and UO_2" a critical mass of 185 kg UO_2 and a SFML of 139 kg UO_2 are obtained. Thus, the KENO/SCALE calculated SFML is in agreement with published data.

If an enrichment of 2.67 wt% uranium-235 (corresponding to burned batch 3 fuel) had been used, a critical mass of approximately 125 kg and a SFML (75% of the critical mass) of approximately 94 kg would have been calculated based on the data in DP-1014 (Clark 1967).

3.2.3 Comparison of the SFML with the Remaining Fuel Quantities

The SFML was compared with the quantity of fuel debris remaining in each area of the TMI-2 facility. This discussion of the comparison between the remaining fuel quantities and the SFML is based on the four major areas discussed previously; the AFHB, the reactor building, the RCS and the reactor vessel.

3.2.3.1 Comparison of SFML with fuel remaining in AFHB

The estimated quantity of fuel in the AFHB as given in the Defueling Completion Report is <17 kg. The largest quantity of fuel in a single cubicle within the AFHB is 4.9 kg in FH109, Spent Fuel Pool "A". The total quantity (<17 kg) is 14% of the SFML calculated by the licensee using 2.24 wt% uranium-235 enrichment and would have been 18% of the SFML if an enrichment of 2.67 wt% uranium-235 had been used.

3.2.3.2 Comparison of SFML with fuel remaining in Reactor Building

The estimated quantity of fuel in the reactor building (not including the RCS or the reactor vessel as given in the Defueling Completion Report is <75 kg. The largest quantity of fuel in a single location is in the sections of the flow distributor which contained incore instrument guide tubes that were bagged and suspended in the "A"

D-ring. Measurements and estimates indicate that there is 24.3 kg of fuel in this location. This is 17% of the SFML calculated by the licensee using 2.24 wt% uranium-235 enrichment and 26% of the SFML using an enrichment of 2.67 wt% uranium-235. Combining fuel from all locations within the reactor building (not including the RCS or reactor vessel) would not exceed either SFML.

3.2.3.3 Comparison of SFML with fuel remaining in RCS

The estimated quantity of fuel in the RCS as given in the Defueling Completion Report is 133 kg. This is below the SFML of 140 kg as calculated by the licensee using an enrichment of 2.24 wt% uranium-235, although above the SFML of 94 kg as calculated using an enrichment of 2.67 wt% uranium-235. However, the largest quantity of fuel in a single location is in the "B" once through steam generator (OTSG) upper tubesheet, where an estimated maximum of 36.3 kg is located. This is 39% of the SFML for 2.67 wt% uranium-235. Although it has been determined that the remaining fuel exists as tightly adherent material and is not readily removable, it should be noted that even if the remaining fuel became mobile and all nonmechanistically accumulated in the lowest point of each half of the RCS, the greatest possible quantity that could accumulate would be less than the 94 kg SFML because of the separation of the two sides (A and B side) of the RCS.

3.2.3.4 Comparison of SFML with fuel remaining in reactor vessel

The estimated quantity of fuel in the reactor vessel is 608.8 kg. This amount is significantly greater than the SFML. For this reason, a separate criticality safety analysis was performed by the licensee. This analysis used in-vessel inspections of debris locations and quantities to develop a specific three-dimensional analytical model of the fuel in the reactor vessel rather than making worst-case assumptions regarding the geometry and reflectivity. A description of the model and of the assumptions used are given in the next section.

3.3 REACTOR VESSEL MODEL

Because the amount of fuel remaining in the reactor vessel is larger than the SFML, a separate criticality safety analysis was performed by the licensee that modeled the location of fuel within the reactor vessel. The results of the licensee's analysis gave a k_{eff} of 0.945, which is below the NRC's acceptance criterion of $k_{eff} = 0.95$ (based on the limit allowed in standard technical specifications for spent fuel storage). The assumptions used in the licensee's analysis are listed and discussed below.

3.3.1 Conservative model of fuel location.

The Defueling Completion Report provided the following estimates for the quantity of debris in the reactor vessel.

	Fuel Remaining (kg of UO_2)
Work Platform Region and Suspended Equipment	30.9
Downcomer Region	178.9
Internals Indexing Fixture Region	4.9
Core Support Shield (CSS) Region	10.7
Upper Core Support Assembly (UCSA) Region	85.3
Lower Core Support Assembly (LCSA) Region	272.7
Bottom Head Region	23.3
Surface Film Deposits	2.1
	608.8

For the purposes of the criticality analysis, fuel debris in the reactor vessel bottom head, the LCSA, and in the core former area (i.e., between the core former baffle plates and the core barrel) in the UCSA were specifically modeled. In other areas of the reactor vessel, the fuel accumulations within the vessel were considered to be too small or were separated from those areas where fuel was located by a far enough distance [the equivalent of approximately 30 cm (12 inches) of water] so as not to cause a reactivity increase due to neutronic interaction between the areas.

Fuel debris was modeled as if it extended radially in from the core barrel to the maximum distance where fuel debris was observed on each LCSA plate. Also, the full periphery of each plate was assumed in the model to be loaded with debris even though some areas did not contain fuel (this latter assumption probably had only a marginal effect on the reactivity of the system). The amount and location of fuel debris that was assumed in this model (2910 kg) was conservative when compared to that which is estimated to remain in the vessel (608.8 kg).

3.3.2 The fuel enrichment was assumed to be 2.96 wt% uranium-235 before burnup.

A pre-burnup enrichment of 2.96 wt% uranium-235 enrichment was assumed (corresponding to the higher enriched batch 3 fuel). The basis for this assumption was discussed previously for the SFML calculation (Section 3.2.1.1). This is a highly conservative assumption.

3.3.3 Fuel burnup effects were considered for the fuel (assumed to be batch 3).

The assumption of fuel burnup is as given previously for the calculation of the SFML and provides a net fuel enrichment of 2.67 wt% uranium-235 for the fuel that is assumed to have an initial enrichment

of 2.96 wt% uranium-235 (batch 3). This is a conservative assumption as discussed for the SFML calculation (Section 3.2.1.2).

3.3.4 The equivalent of standard, full sized fuel pellets were used for the fuel particle size.

The equivalent of standard, full sized fuel pellets were used for the fuel particle size, as discussed previously for the calculation of the SFML (Section 3.2.1.3).

3.3.5 No credit was taken for the presence of structural and solid poison materials existing in the fuel debris.

As discussed in Section 3.2.1.4, the only poisons assumed during the analysis are those associated with the burnup of UO_2. Thus, impurities that have been identified as existing with the fuel (for example boron, iron, zirconium, and cadmium) were not included, thus providing a conservative result. In addition, no credit was taken for the licensee's plans to add a stable and insoluble neutron poison material to the bottom head of the reactor vessel to provide an added margin of assurance.

3.3.6 The model included moderation with unborated water

Full water reflection was assumed in the analysis and fuel was treated as a fuel-water only mixture (except for the lower core support assembly (LCSA) plates in which case the holes in these plates were assumed to contain fuel that is homogeneously mixed with the plate material). The presence of boron (which is a neutron poison) in the water was ignored in the criticality analysis. No credit was taken for the licensee's plans to drain the water from the reactor vessel following the defueling operations thus leaving the reactor vessel without a moderating medium.

3.3.7 The computer code KENO V.a was used and an uncertainty of 2.5% Δk was added to the calculated k_{eff}.

The computational bias of 2.5% Δk is added to the calculated k_{eff} as discussed for calculation of the SFML (Section 3.2.1.6).

3.4 NEUTRONIC COUPLING OF IN- AND EX-VESSEL FUEL

The licensee's model did not account for the potential of neutronic coupling of the core debris within the vessel and the debris located in other areas of the facility (the RCS, the reactor building or in the AFHB). This is an appropriate assumption because the debris in the vessel is well separated from the debris in other locations and no identifiable methods exist for transport of the fuel into or out of the vessel.

3.5 ACCIDENT ANALYSIS

Given the type and physical condition of the fissile material present, criticality would only be possible if this material were to accumulate in greater quantities at some location. All areas outside of the reactor vessel contain less than the safe fuel mass limit, and intermixing of the fuel debris in these areas could only be accomplished by design. The only area where fuel might be able to accumulate is in the lower head of the reactor vessel. The licensee's analysis of the result of a redistribution of the available material into the lower head assumed that 350 kg are available for redistribution (including all loose, fine, granular debris and surface films). The results of the analysis, which assumed a full water reflected slab of material, gave a k_{eff} of 0.913.

This is a conservative analysis, since much of this debris is in the annular gap, and thus would not shake down into the lower head. In addition, the absence of water, and the presence of neutron poisons including; 1) impurities in the residual fuel, 2) boron in any remaining water, 3) structural material and 4) an insoluble poison that will be added by the licensee to the bottom head of the reactor vessel following draining of the RCS, will further ensure the lack of a criticality event even under accident conditions.

4.0 CONCLUSIONS

Based on our analysis of the licensee's Defueling Completion Report, we conclude that there is no potential for criticality in the fuel remaining in the TMI-2 facility during either normal or accident conditions. Although two of the assumptions used by the licensee to calculate the safe fuel mass limit [1) an enrichment after burnup of 2.24 wt% uranium-235 and 2) the k_{eff} limit criterion of 0.99 rather than 0.95] are not as conservative as the data appears to warrant, our review indicates that use of the more conservative assumptions does not signal a potential for a criticality. The conservatisms built into the model, as well as the safeguards that the licensee will enact (including removal of water and addition of a neutron poison into the vessel) would further preclude a criticality.

5.0 REFERENCES

Barner, J.O. 1985. Characterization of LWR spent fuel, MCC-Approved Testing Material - ATM-101. PNL-5109, Rev. 1. Pacific Northwest Laboratory, Richland, Washington.

Clark, H.K. 1967. Critical and Safe Masses and Dimensions of lattices of U and UO_2. DP-1014. E.I. Du Pont DeNemours and Company, Savannah River Laboratory, Aiken, SC.

Attachment 3

April 26, 1990

Docket No. 50-320

DISTRIBUTION
Docket File
SNorris MMasnik
PDI-4 (Memo file)
LThonus
FYoung RBrady

MEMORANDUM FOR: John F. Stolz, Director
Project Directorate I-4
Division of Reactor Projects - I/II

FROM: Lee H. Thonus, Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II

Francis I. Young, Senior Resident Inspector
Three Mile Island Site
Region I

Robert R. Brady, Resident Inspector
Three Mile Island Site
Region I

SUBJECT: FUEL CANISTER VERIFICATION AT TMI-2

On March 23, 1990, F. I. Young and R. R. Brady performed an inspection of the TMI-2 reactor building. As part of the inspection they verified that no canisters containing core material remained in the reactor building. This satisfies condition "C" of Technical Specification Table 1.1 for transition from Mode 1 to Mode 2.

On April 20, 1990, L. H. Thonus and R. R. Brady inspected the spent fuel storage pool "A", the truck bay, and the rail siding. The spent fuel pool "A" contained three "dummy" canisters (used to check out the fuel transfer system) and one empty filter canister. These areas represent the pathway for canisters shipped offsite and the potential storage locations for canisters. The inspection team verified that no canisters containing core material remained in these areas. Therefore, no canisters containing core material are stored at the TMI-2 site. This fulfills the requirement for TMI-2 to transition from Mode 2 to Mode 3.

These inspection results will be part of a forthcoming Inspection Report to be issued by Region I regarding the transition of TMI-2 from Mode 1 through Mode 3.

signed by M. Masnik for
Lee H. Thonus, Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II

signed by M. Masnik for
Francis I. Young, Senior Resident Inspector
Three Mile Island Site
Region I

signed by M. Masnik for
Robert R. Brady, Resident Inspector
Three Mile Island Site
Region I

cc: T. Sezmanski, NRR/LOLB
E. Wenzinger, Region I

LA:PDI-4	PM:PDI-4	PM:PDI-4	SRI	RI:REGION I	PD:PDI-4
	MMasnik for	MMasnik	MTM for	MMasnik for	
SNorris	LThonus:lm	MMasnik	FYoung	RBrady	JSTOLZ
4/26/90	4/26/90	4/26/90	4/26/90	4/26/90	4/26/90

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Enclosure 3

Cleanup Schedule for Three Mile Island, Unit 2

Completion of defueling	January 30, 1990
NRC-sponsored lower head sampling	January 30 - March 1, 1990
Submission of defueling completion report	February 26, 1990
Completion of offsite fuel shipment	April 15, 1990
Transition from Mode 1 to Mode 2	April 26, 1990
AGW evaporation	October 1990 - February 1992
Issuance of staff SER for PDMS	Fall 1990
NRC-sponsored fuel measurement program	Fall 1990
RCS drawdown	Fall 1990
Completion of the licensee's fuel measurement program	Early 1991
Final decontamination of reactor building	December 1991
TMI-2 ready for long-term storage (PDMS)	December 1992

Enclosure 4

Post-Defueling Monitored Storage (PDMS) of Three Mile Island, Unit 2

At a 1985 TMI-2 Advisory Panel Meeting a representative of the Commonwealth of Pennsylvania, and a member of the TMI-2 Advisory Panel, proposed that GPU Nuclear Corporation (the licensee) should consider a storage period after defueling and before large scale decontamination of the reactor building basement. The Panel members concern was the possibility of avoiding a large worker dose associated with this decontamination effort. The licensee pursued this approach and on December 2, 1986 submitted a plan for plant conditions during what the licensee termed as Post Defueling Monitored Storage or PDMS.

Based on the licensee's December 2, 1986 submittal the staff issued Draft Supplement 3 to the TMI-2 Programmatic Environmental Impact Statement in April 1988. Comments on the Draft Supplement were received from 23 Federal and State agencies or members of the public.

The licensee has kept the Commission informed of their plans to place the facility into PDMS at a number of Commission briefings held in the Washington, D.C. area. The licensee first informed the Commission of their plan to store the facility after partial cleanup in a January 14, 1986 Commission meeting. At the February 13, 1987 Commission meeting the licensee proposed the term PDMS for the storage period. On March 17, 1988 the licensee again discussed PDMS with the Commission.

On August 16, 1988, the licensee submitted a request for a license amendment that would allow for long-term storage of the facility when offsite waste shipments were completed and the plant reached a safe and stable condition. The August 16, 1988, submittal stated that even though the amendment would result in a "possession only" license, this request did not represent a decision by the licensee to decommission the plant and should not be construed as permanent cessation of operations.

Subsequent to the August 16, 1988 submittal, the licensee informed the staff (Attachment 1) that the licensee's current energy supply plans do not reflect the return to service of TMI-2 and that before expiration of the current TMI-2 license, a decision will be made to decommission the plant without further operation. Decommissioning of Unit 2 will probably occur when Unit 1 is decommissioned.

The decommissioning rule links a "possession only" license to permanent cessation of operations and the submittal of a proposed decommissioning plan within 2 years of ceasing operations. Since the licensee's decommissioning plans are deferred, and they have stated that its submittal does not represent a decision to permanently cease operations the regulations pertaining to decommissioning are not applicable at this time. The licensee would, however, be required to submit a decommissioning funding plan by July 27, 1990 as required by 50.75(b) and 50.33(k)(2).

On September 22, 1989, the staff issued Final Supplement 3 to the Programmatic Environmental Impact Statement (PEIS) for TMI-2. Supplement 3 evaluates the environmental impacts associated with the licensee's proposal for long-term storage of the facility, as well as a number of alternatives. The staff found the licensee's proposal, as well as a number of the alternatives, environmentally acceptable and that none of the alternatives was obviously superior to the licensee's proposal.

The staff and its contractor, Pacific Northwest Laboratory (PNL), are presently reviewing the licensee's safety analysis report for PDMS, and the staff expects to issue a safety evaluation report (SER) in the fall of 1990. The review focuses on fire protection, ventilation, and containment integrity. The issue of a potential inadvertent criticality occurring after the conclusion of defueling will have already been resolved as part of the staff's review of the defueling completion report. PEIS Supplement 3 and the staff's SER will form the basis for the staff's position on the acceptability of PDMS.

Actual implementation of PDMS will require a license amendment to change the technical specifications (TS) for the facility. The licensee's August 16, 1988, submittal includes a request to amend the TS to allow for PDMS. Attachment 2 provides a listing of the TS proposed by the licensee during PDMS. The proposed TS include only administrative requirements and the requirement to maintain containment isolation. Since the licensee plans to revise the PDMS amendment, the staff has delayed issuance of a notice on the licensee's August 16, 1988, submittal.

Based on the need for a detailed technical review, principally involving the assessment of remaining plant operating systems needed for PDMS, and the considerable public interest, the staff plans to issue a notice of opportunity for a prior hearing on the amendment request. If a hearing is required on this issue, it is estimated that the proceeding will take approximately 24 months. An Atomic Safety and Licensing Board Panel decision on PDMS would not be expected until spring of 1992.

The licensee has informally requested that the notice of opportunity for a prior hearing on the amendment request and the issuance of the staff's SER on PDMS be delayed until the end of the year. The licensee plans to update the SAR on PDMS, take some additional fuel measurements which further refines earlier estimates and plans to complete some additional decontamination necessary for PDMS. The staff finds the licensee's application sufficiently complete at this time to go forward with issuance of the notice and the SER in the next couple of months.

The staff expects intervention on this amendment. Intervention would probably focus on why the cleanup has not progressed further or has not been completed, and why this facility is to be placed in a special category and not decommissioned immediately at the conclusion of defueling.

Decommissioning in the public's view is dismantlement, and the public fails to recognize that SAFSTOR is one of the acceptable decommissioning options. For all practical purposes, PDMS is SAFSTOR; the only difference is the name and the fact that the licensee has not announced that it has "permanently ceased operations."

Assuming approximately 24 months for the hearing process, a decision on PDMS is not expected until 1992. By the time the licensee will be able to implement PDMS, most of the reduction in TS requirements will have taken place during the mode changes (see Attachment 2). The impact of any hearing on implementing PDMS should not impose a significant hardship on the licensee regarding resources needed to maintain TMI-2 as there is little difference between Mode 3 and PDMS. The additional reductions requested in the August 16, 1988 license amendment request for PDMS are primarily some reductions in surveillance requirements during the period of storage.

GPU Nuclear

P.R. Clark
President

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June 23, 1989

Mr. Victor Stello, Jr.
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Sir:

Three Mile Island Nuclear Station, Unit 2 (TMI-2)
Operating License No. DPR-73
Docket No. 50-320

This responds to your letter of May 15, 1989, in which you asked GPU Nuclear to advise you about our future plans for the TMI-2 facility commencing with Post-Defueling Monitored Storage (PDMS). Your letter states that the purpose of the request is to provide a better understanding of the basis for our Decommissioning Funding Plan. Pursuant to 10 CFR 50.33(k)(2), that plan is due to be filed with the NRC on or before July 26, 1990.

GPU Nuclear is proceeding to complete the cleanup of TMI-2 to achieve a safe, stable, and secure condition which presents no risk to the environment or to public health and safety for its duration. In March 1987 an Environmental Evaluation of the monitored storage condition was forwarded to the staff, and in August 1988 a detailed Safety Analysis Report with accompanying license amendment request, which further described the monitored storage condition, was submitted for NRC review and approval. The thrust of our proposal is that TMI-2 will have been defueled and cleaned up to the degree that it poses no risk to public health and safety. A period of monitored storage would significantly reduce occupational exposure (by 5000 to 7000 person-rem) at the time of eventual decommissioning and would allow simultaneous decommissioning with TMI-1. Our proposal is that monitored storage of TMI-2 would not extend beyond decommissioning of TMI-1.

As reported by Mr. William G. Kuhns and affirmed by the current Chairman of General Public Utilities Corporation, Mr. Standley H. Hoch, at the recent April 1989 meeting with the Commissioners, the GPU System's present energy supply plans do not reflect the return to service of TMI-2. Mr. Kuhns also noted, "As all our efforts are devoted to completing the cleanup, we are not studying the details of eventual disposition of the plant at this time."

GPU Nuclear understands and accepts responsibility as a licensee to ensure proper disposition of the plant and eventual termination of the license. In our August 5, 1988 letter to the NRC, we stated that we understand that the NRC's rule for decommissioning nuclear facilities applies to TMI-2 and would cover all activities involved in decommissioning the plant starting from monitored storage conditions. Therefore, we will provide by July 26, 1990 the decommissioning funding plan and certification required by that rule for TMI-2. We expect that funding plan to be based on the assumption that prior to the end of the current licensed life a decision will be made to decommission the plant without further operation. The plan will recognize the unique conditions of TMI-2 which, we anticipate, will require funds substantially greater than those required to decommission TMI-1 and, therefore, will exceed the minimum amount required by the rule.

We trust this is responsive to your request.

Sincerely,

P.R. Clark

P. R. Clark
President

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ATTACHMENT 2

TECHNICAL SPECIFICATION COMPARISON MATRIX

TECH. SPEC. TITLE	TECH. SPEC. NUMBERS	MODE 1 CURRENT STATUS	MODE 2 POST-DEFUELING	MODE 3 DEFUELING AND FUEL SHIPMENT COMPLETE	MODE 4 POST-DEFUELING MONITORED STORAGE (PDMS) TECH. SPECS.	COMMENTS
APPENDIX A TECH. SPECS.						
1. Borated Cooling Water Injection and Boron Concentration - Reactor Coolant System	3.1.1.1 & 3.1.1.2	X				
2. Minimum Boron Concentration - Fuel Transfer Canal	3.1.1.3	X				
3. Minimum Boron Concentration - Spent Fuel Pool	3.1.1.4	X	X			
4. Neutron Monitoring Instrumentation	3.3.1.1	X				
5. Radiation Monitoring	3.3.3.1	X	X	X		As required in Table 4.3-3 of the Recovery Operations Plan. Additional radiation monitoring requirements are addressed in the Appendix B Tech. Specs.
6. Meteorological Instrumentation	3.3.4.4	X	X			
7. Essential Parameters Monitoring Instrumentation	3.3.3.5	X				
8. Chlorine Detection System	3.3.3.7	X				
9. Fire Detection Instrumentation	3.3.3.8	X	X	X		Sect. 7.2.2.2.b of the PDMS SAR states that the zone detection system originally provided at TMI-2 will remain operational throughout operational areas of the plant during PDMS (Mode 4). A common visual and audible alarm will be installed to provide monitoring capabilities and remote fire alarm indication in TMI-1.
10. Reactor Vessel Water Level Monitoring	3.4.2	X				
11. Pressure/Temperature Limits RCS	3.4.9	X				

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12. Communications - Control Room	3.5.1	X				
13. Containment Integrity (Double Valve and Airlock Door Isolation)	3.6.1.1 & 3.6.1.3	X				
14. Containment Isolation (Single Valve and Airlock Door Isolation)	3.6.1.2 & 3.6.1.6		X	X	X	
15. Reactor Building Internal Pressure Limits	3.6.1.4	X	X	X		During PDMS (Mode 4), active pressure control of the RB will not be maintained. A Containment Atmospheric Breather will be added to provide passive pressure control of the RB relative to ambient atmospheric pressure (Ref. Sect. 1.2.3.3 of the PDMS SAR).
16. Reactor Building Air Temperature	3.6.1.5	X				
17. Containment Purge Exhaust System Operability	3.6.3.1	X	X	X		Sect. 7.2.1.3 of the PDMS SAR stated that the Containment Purge Exhaust System will be maintained in an operable condition to support PDMS (Mode 4) activities (e.g., surveillance entries, maintenance).
18. Flood Protection	3.7.6.1	X	X	X		Sect. 1.4 of the PDMS SAR states that existing unit flood protection capabilities will be maintained for PDMS (Mode 4). Specific design features are: -Reactor Bldg- No external openings in the RB below 305' elev. (ground level). -FH Bldg- No external openings in TMI-2 FHB that require flood protection. Railroad door in the TMI-1 porportion of the FHB is designed to be watertight. -Control Bldg- Flood panels and doors provided. -Aux Bldg- Shield door is watertight. -DG Bldg- Flood panels watertight.

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19. Control Room Emergency Air Cleanup System	3.7.7.1	X				
20. Sealed Source Integrity	3.7.9.1	X	X	X		The PDMS SAR does not address sealed sources. There may be a limited need for sealed sources during PDMS (Mode 4) for calibration of radiation monitors. Site procedures will address the use and calibration of sealed sources, if necessary.
21. Fire Suppression Water System	3.7.10.1	X	X	X		NRC approval of TSCR 57 will delete the TMI-2 required water sources. Sect. 7.2.2.2.a of the PDMS SAR stated that the yard fire main will be maintained pressurized using the station fire pumps in TMI-1. The TMI-2 Fire Protection System draws its supply water from the tie-in to the yard main.
22. Deluge/Sprinkler Systems	3.7.10.2	X	X	X		NRC approval of TSCR 57 will delete this specification. Sect. 7.2.2.2 of the PDMS SAR states that the fire service loop which runs through the Diesel Generator Bldg, AFHB, and Control Bldg will be isolated during PDMS (Mode 4). Drain valves will be installed and capability will exist to quickly energize the systems. The deluge systems for the Aux. Transformers and east wall curtain are maintained Turbine Building.
23. Halon System	3.7.10.3	X				NRC approval of TSCR 57 will delete this specification. Halon system will be deactivated during PDMS (Mode 4).
24. Fire Hose Stations	3.7.10.4	X	X	X		NRC approval of TSCR 57 will reduce the required fire hose stations to the Reactor Bldg only. Sect. 7.2.2.1 of the PDMS SAR states that hose reel and hose cabinet stations shall be provided in areas of the facility where systems or equipment are maintained operational for PDMS (Mode 4).
25. Penetration Fire Barriers	3.7.1.1	X	X	X		NRC approval of TSCR 57 will delete this specification. PDMS SAR does not address this area. Fire protection organization does not plan to perform any routine surveillance or penetration fire barriers during PDMS (Mode 4).

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APPENDIX A TECH. SPECS.						
26. Electrical Power System	3.8	X				
27. Spent Fuel Pool Water Level Monitoring	3.9.1 & 3.9.2	X	X			
28. Fuel Transfer Canal Water Level Monitoring	3.9.3 & 3.9.4	X				
29. Fuel Handling Building & Auxiliary Building Ventilation Systems Operability	3.9.12.1 & 3.9.12.2	X	X	X		Sects. 7.2.6.1 and 7.2.6.2 of the PDMS SAR state that the FHB and Aux Bldg ventilation systems will be maintained in an operational status to support PDMS (Mode 4) activities.
30. Accident-Generated Water	3.9.13	X	X	X	X	Procedures require NRC approval. Sect. 1.1.3 of the PDMS SAR states that because the issue of disposal of AGW is a separate, well-bounded activity, it is not discussed in the PDMS SAR. Additionally, the PDMS SAR states that disposal of AGW will likely extend into PDMS (Mode 4) but will not materially interefere with maintaining the TMI-2 plant in a safe, monitored PDMS condition.
31. Heavy Load Handling Restrictions in Reactor Building	3.10.1	X				
32. Heavy Load Handling Restrictions in Fuel Handling Building	3.10.2	X	X			
33. Responsibility	6.1.1	X	X	X	X	
34. GPUNC Organization	6.2.1	X	X	X	X	Reduced in scope for PDMS (Mode 4). In addition, the Tech. Specs. refer to the PDMS SAR org. chart rather than a separate organization plan.
35. TMI-2 Organization	6.2.2					
a. Minimum Shift Crew Composition	6.2.2.a	X				
b. Licensed Operator in Control Room When Fuel is in Reactor	6.2.2.b	X				

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c. Individual Qualified in Radiation Protection Procedures When Fuel is in Reactor	6.2.2.c	X				
d. Site Fire Brigade	6.2.2.d	X	X	X		Unit 1 is presently manning the Site Fire Brigade and will continue to do so during PDMS (Mode 4).
e. Individual Qualified in Radiation Protection Procedures When Radwaste Mangement Activities are in Progress	6.2.2.e	X	X	X	X	Figure 10.5-2 of the PDMS SAR notes that during PDMS (Mode 4), TMI-2 will continue to have a department dedicated to Waste Management/Decontamination. It is currently planned that radiation protection personnel will be supplied on a site basis during PDMS (Mode 4).
36. Unit Staff Qualifications	6.3	X	X	X	X	
37. Training	6.4	X	X	X	X	Training program requirement for Fire Brigade is is deleted during PDMS (Mode 4).
38. Technical Review and Control	6.5.1	X	X	X	X	
39. Independent Review - Function	6.5.2.1 - 6.5.2.7	X	X	X	X	
40. Independent Safety Reviewers (ISRs)	6.5.2.8			X	X	This specification will become effective upon NRC approval of TSCR 60; the ISRs will perform similar functions to SRG but are not a full-time dedicated group.
41. Audits	6.5.3	X	X	X	X	The number of required audits are reduced during PDMS (Mode 4).
42. Minimum Staffing Requirements for the Safety Review Group (SRG)	6.5.4	X	X			NRC approval of TSCR 60 will reduce the minimum staffing requirements by two (2) personnel in Mode 2 and delete the SRG in Mode 3.
43. Reportable Events	6.6.1	X	X	X		During PDMS (Mode 4), violation of Tech. Spec. Action Statements will be included as part of the annual report.

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APPENDIX A TECH. SPECS.						
44. Procedures	6.8	X	X	X	X	Reduced in scope for PDMS (Mode 4.
45. Report Requirements	6.9	X	X	X	X	
46. Record Retention	6.10	X	X	X	X	
47. Radiation Protection Program	6.11	X	X	X	X	
48. High Radiation Area	6.12	X	X	X	X	
APPENDIX B TECH. SPECS.						
1. Liquid Effluents	2.1.1	X	X	X	X	Reduced in scope for PDMS (Mode 4).
2. Gaseous Effluents	2.1.2	X	X	X	X	Reduced in scope for PDMS (Mode 4).
3. Radioactive Gaseous Effluent Monitoring Instrumentation	2.1.3	X	X	X		This sect. applies to the EPICOR II ventilation system monitor only which is not a PDMS (Mode 4) system.
4. Environmental Monitoring (Non-radiological) Requirements	3.1	X	X	X		Neither the PDMS SAR nor the Environmental Evaluation proposes to continue this program.
5. Radiological Environmental Monitoring Program (REMP) Requirements	3.2	X	X	X		Though not part of the PDMS (Mode 4) Tech. Specs., this program will continue as discussed in Sect. 3.5 of the PDMS Environmental Evaluation.

Lower Head Sampling Program

On March 1, 1990, the Vessel Inspection Program (VIP) sponsored by the NRC's Office of Nuclear Regulatory Research and the Organization of Economic Cooperation and Development (OECD) was completed. A total of 14 "boat" samples was obtained from the reactor vessel lower head. Three of these included the base of the incore instrument penetrations (IIP). Additionally, 14 IIPs were cut off 1.5 to 2 inches above the lower head and obtained as samples. Two incore instrument guide tubes were cut free from the flow distributor head as samples.

The samples were shipped to Argonne National Laboratory (ANL) for analysis and distribution to the sponsoring OECD nations. The staff expects preliminary results within the next several months.