JUN 2 9 1988

F. R. Standerfer, Director
Three Mile Island Unit 2
GPU Nuclear Corporation
P.O. Box 480
Middletown, PA 17057

Dear Mr. Standerfer:

Enclosed are responses to questions raised at the May 26, 1988 TMI2 Advisory Panel Meeting. Responses have been prepared to written questions submitted by TMIA and SVA. Questions raised during the meeting are either answered in the enclosure or referenced by page number back to the transcript for the May 26, 1988 meeting.

I hope to see you at our July 14, 1988 meeting in Harrisburg, PA.

Sincerely,

ORIGINAL SIGNED BY

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

Enclosure:
As stated

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QUESTIONS RAISED AT THE MAY 26, 1988 ADVISORY PANEL MEETING, HARRSIBURG, PA

Susquehanna Valley Alliance

1. Due to the uncertainties of radionuclide dispersion and deposition following the accident, upon what information is Table 2.4 based?

   Response:

   Because refined modeling methods are not available for accurately analyzing the transport and deposition of the fragmentation debris, or the leaching of soluble materials from the damaged core, a set of assumptions was made regarding the dispersion and deposition of radionuclides in the TMI-2 facility. These assumptions were based on information available from fuel measurements and contamination measurements throughout the reactor building, as well as on the chemical and physical state of the radionuclides. All assumptions were chosen to ensure that the amount of activity estimated to be in any location either meets or exceeds the amount actually measured in that location. The assumptions are outlined in Section 2.2 of NUREG-0683, Draft Supplement No. 3 on pages 2.21 through 2.31.

2. By the time of PDMS, will we know the condition of the containment and the damage to it caused by the accident? How will this information be made available to the Public?

   Response:

   There has been no evidence of any damage to the containment building that would result in any compromise in its ability to contain radiation during PDMS. Worker access is available above the 305 ft elevation, and no signs of containment degradation have been observed. Video access of the 282 ft elevation (the reactor building basement) has not disclosed any damage to the containment building.

3. While Unit 2 is in PDMS, what research will continue which relates to the reactor?

   Response:

   The NRC has no plans for additional research directly related to TMI-2 during the proposed post-defueling monitored storage period.

4. Explain the rationale for delaying clean-up. Delay will have no effect on the long-lived radionuclides. Is the delay then for reasons of technological advances?

Response:

The NRC, in its role as a regulatory agency is evaluating GPU Nuclear's proposal to place the TMI-2 facility in post-defueling monitored storage. According to the licensee's Technical Plan, (TMI-2 Cleanup Program Post-Defueling Monitored Storage (TPO/TMI-188 Rev. 0), January 1987):

> "A monitored storage period following completion of the current cleanup program is beneficial for several reasons.
>
> Occupational dose in the plant will be reduced during monitored storage due to the natural decay of radioactive contamination. Over an extended period, levels for the dominant isotopes (Sr-90, Cs-137) could be reduced by as much as a factor of 2 [more likely only two-thirds]. The occupational dose in radiation zones would be reduced proportionately.
>
> The monitored storage period allows time for continued development of decontamination technology so that the most effective and efficient techniques may be applied. Further reduction in occupational exposures would be achieved through use of advanced robotic technology, automatic cleaning and chemical cleaning techniques, and advanced waste treatment methods.
>
> This monitored storage period also allows for resolution of the current limitation on national waste disposal capabilities so that selection of processes may be less dependent on waste volume production. The result may be further reductions in occupational dose required to accomplish specific tasks."

5. How will the number of entries be determined during PDMS?

Response:

The licensee indicated in "TMI-2 Cleanup Program Post-defueling Monitored Storage - Technical Plan" (TPO/TMI-188, January 1987) that entries to the reactor building and auxiliary and fuel handling building would be conducted for purposes of visual inspection, radiation survey, and recording of plant conditions (TPO/TMI-188). Table 2.3 of that document (Table 3.2 of NUREG-0683, Draft Supplement 3) lists the anticipated schedule for initial PDMS monitoring/inspections. The number of entries will be greatest early in PDMS. Although the licensee's plan for the initial frequency of entries is monthly (12 times per year), the licensee indicates that they anticipate "that the initial frequency will decrease (e.g., quarterly) based on an evaluation of data accumulated during the initial period." (TPO/TMI-188, p.20).

6. Upon what findings and/or studies does the NRC base its assumption that the activity in the top 1/2" of the wall becomes available for resuspension? What allowances are made for the fact that the walls might crumble due to stress from age and clean-up activities already undertaken?

Response:

Page 3.12 (section 3.2.2.1) of Draft Supplement 3 contains the assumption that the activity in the first 1/2 inch (1.3 centimeters) of the concrete block becomes available for resuspension after the structure has dried for a period of time. This assumption is based on a study by Arora and Dayal (1986) as referenced in NUREG-0683, Draft Supplement 3. This study indicated that for cesium in a concrete solid, the cesium leach rates were greater when the wet periods were interspersed with dry periods, than when the concrete solid was continuously saturated. The observed enhancement in cesium release with increasing length of dry period is believed to be a result of replenishment of the surface with cesium, migrating from the sub-surface zones during dry periods. In Draft Supplement 3 this phenomenon was bounded, by assuming that up to one-eighth of the radioactive material in the concrete blockwall would migrate to the surface and be available for suspension into the atmosphere. This number is at least several times greater than the amount of radioactive material that is expected to be available for resuspension from the concrete block wall.

The reactor building is a reinforced concrete structure composed of a cylindrical wall with a flat foundation mat and a dome roof. The wall thickness of the cylindrical wall is 4 feet and the thickness of the dome is 3 feet, 6 inches. The foundation mat on bearing rock is 11 feet, 6 inches thick with a 2-foot thick concrete slab above the base liner plate. The inside surface of the reactor building is lined with a carbon steel liner with a nominal thickness of 3/8 inches for the cylinder, 1/2 inch for the dome and 1/4 inch for the base.

The TMI-2 facility was designed and constructed for a 40-year lifetime from the start of construction (beginning late in 1969) and the staff, in Draft Supplement 3, assumed post-defueling monitored storage period would be complete in 2009 (a period of 40 years from the start of construction). The cleanup activities that have occurred or are being proposed for the period before PDMS are relatively nondestructive in nature. The environment to which the walls of the containment have been exposed to since the accident would not cause any significant degradation of the concrete. The NRC staff did not consider the crumbling of walls due to stress from age or cleanup activities credible.

7. What is 10% of each activation product? Upon what information or studies do you make the assumption that 10% of the activation products will remain in the reactor building at the end of defueling? (EIS page 2.27, 2.2.1).

Response:

Table 2.4 lists the quantity of activation products that were assumed to be present in the facility after defueling. Activation products listed in Table 2.4 include manganese-54, iron-55, cobalt-60 and nickel-63. Carbon-14 is also formed by activation, although a small quantity is formed by fission. Carbon-14 was treated as a fission product because it has been detected in the accident-generated water and is therefore soluble in water and assumed to be distributed in the same manner that was assumed for some of the fission products. Tritium is also produced by activation although over 90% of the tritium produced is from ternary fission.

The amount of activation products present at the time of the accident was determined using the ORIGEN-2 computer codes. The amount that would have been available at the beginning of PDMS in the absence of any cleanup or defueling operations was calculated by decay correcting the amount present after the accident. The assumption was made that only a small portion of the activation products were removed by sampling or defueling. Of the total quantity of activation products estimated to be present at the end of defueling, 10 percent was assumed to be in a form that would allow for dispersal and could contribute to an offsite dose. The assumption was made that the remaining 90 percent of the activation products were either shipped offsite or were part of the stainless steel of the primary system and therefore unavailable for dispersal.

8. The water which will leak into the system has been determined to be 5000 gallons per year. Explain why this amount is so much less than the inleakage for this past 9 years.

Response:

According to the environmental evaluation (Letter from F. R. Standerfer to the NRC, March 11, 1987. Subject: Environmental Ev~luation for TMI-2 Post-Defueling Monitored Storage. 4410-87-L-0025, Document ID 0161P) inleakage of groundwater and precipitation are anticipated to be the major sources of liquids during PDMS. The licensee estimated, based on experience to date and on the anticipated lower frequency of maintenance during PDMS, an annual inleakage of 5000 gallons. Water inleakage currently occurs in the following areas of the plant and is collected as indicated:

1) Fire service penetration; east wall of turbine building at the 300-ft elevation. Drainage is to the turbine building sump, water treatment sump, or the condensate regeneration polisher sump.

2) Building joint; between the service building and air intake tunnel. This area does not have sump drainage. It is pumped periodically, as necessary, to remove inleakage.
3) Construction joint; basement of the auxiliary building. Drainage is to the auxiliary building sump.
4) Electrical penetration; southwest corner of the control building area at the 281' elevation. Drainage is to the control building area sump.

Reference (Letter, Standerfer to the NRC, June 23, 1987. Subject Post-defueling monitored storage environmental evaluation. 4410-87-L-0093, Document ID 0194P. - Referenced in Draft Supplement 3 and available in Public Reading Room).

No inleakage is expected into the reactor building.

The expected annual inleakage of 5000 gallons is much less than the amount (approximately 264,000 gallons) of water that flowed into the reactor building basement during the two years following the accident. The sources of this water included the primary coolant, water from the reactor building spray system and from river water inleakage from the building air coolers. However, the reactor coolant system will be drained before PDMS begins, the building air cooler system has already been drained and the closed loop (does not use water from the river) cooling system currently in use will be deactivated and drained before PDMS begins.

9. Page 3.31, Section 3.3.1.1. Explain those measurements which are being presently undertaken? What is being measured? In what manner will the results affect decisions about RCS decontamination and the future of the facility?

Response:

The statement in question refers to the measurement of radioactive material located in the reactor coolant system. The amount of radioactive material including the amount of fuel debris is being measured in all accessible locations of the reactor coolant system. The methods that will be used during the decontamination of the reactor coolant system will depend in part on the amount of radioactive material present in the reactor coolant system and its precise location in the system. For instance, those areas with little or no contamination will require very minor amounts of decontamination, while decontamination efforts in areas that contain large amounts of radioactive material will be more extensive.

The draft supplement on page 3.31 indicated that the selection of methods and processes for additional reactor coolant system decontamination is expected to depend on the future disposition of the facility and on measurements being made at the present time. We did not intend to indicate that the results of the measurements would affect decisions on the future of the facility.

10. What would preclude the use of the AGW to clean the RCS?

Response:

No action other than disposal of the accident-generated water would preclude its use during the decontamination of the reactor coolant system. For the evaluation in Draft Supplement 3 to the PEIS, it was assumed that the accident-generated water would be processed and removed from the reactor building and the auxiliary and fuel-handling building prior to the initiation of immediate cleanup or post-defueling monitored storage.

11. Will the water used for further clean-up contain chemicals? How will these be removed from the water before the water is released to our drinking water supply?

Response:

Water that is used during decontamination and cleanup processes is routinely run through ion-exchange systems (the submerged demineralizer system and the EPICOR II system are ion-exchange systems) if necessary to filter any radioactive material and chemicals that may be present. Any water released to the Susquehanna River or to any other drinking water supply would have to meet the licensee's technical specifications as well as the conditions of the National Pollutant Discharge Elimination System (NPDES) permit issued by the Commonwealth of Pennsylvania, Department of Environmental Resources (PaDER).

12. Page 3.32, Section 3.3.1.2. When do you expect the radiation doses to be low enough to permit entry into the basement for complete clean-up? If they are presently too high to permit entry, does this not rule out the possibility of immediate clean-up as an alternative to be considered?

Response:

Entry into the basement would most likely not be considered in areas where the dose rate remained much above 1 R/hr, although, even at these radiation levels, a worker could be allowed to work for a short period of time. High dose rates, however, do not preclude the possibility of cleaning the basement, or the possibility of the immediate cleanup alternative. Dose reduction efforts are currently occurring in the reactor building basement. These dose reduction efforts, including scabbling of the walls with robots and construction of a manifold for waterflow to leach activity from the concrete block wall, are described in Draft Supplement 3, in section 2.1.1, pages 2.9 to 2.11.

13. How can the impact of the waste after disposal at either a regional or other site be considered outside the scope of this EIS? Delaying clean-up has a major impact on the final resting place for the waste from TMI, since the State of Pennsylvania is presently in the process of developing a site.

Response:

The environmental impact of waste disposal at a commercial low-level waste disposal site is the subject of an environmental evaluation specific to the chosen site, which must be completed before the site can be licensed. Waste streams outside those evaluated during the sites environmental evaluation will not be allowed for burial. The environmental evaluation for a regional burial site must be specific to the environmental characteristics of the site, and must also address all types of wastes that will be accepted into it, including wastes from hospitals and university research labs. Wastes from TMI-2 will not be accepted at a regional site until the site is licensed.

14. Page 3.19, footnote a. What are the precautions to be taken to ensure that criticality would not occur?

Response:

A variety of precautions are available for use during the cleanup program to ensure that criticality will not occur. These include ensuring that the small quantity of fuel debris remaining after the current defueling efforts is not available in large enough quantities to create any possibility of a criticality. 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 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 such that an inadvertent criticality is precluded.

15. Between entries, how will the Licensee know that criticality has not occurred?

Response:

Prior to entering PDMS, steps will be taken to ensure that a criticality event is not a credible event (see response to question 14 above). Most of the fuel debris remaining in the TMI-2 facility following the current defueling effort would be sealed in piping or enclosed in components. Measurements will be made by the licensee and verified by the NRC and their contractors to ensure that the amount of fuel debris in a given area will not be large enough to cause a criticality. During PDMS the licensee does not plan to maintain monitoring activities that are specific to identifying an inadvertent criticality event in containment.

16. During entries, how will the workers know that criticality is not occurring?

Response:

See response to Question No. 15 above. Furthermore, workers will carry radiation survey instruments.

17. By what means will the Licensee determine the amount of radioactivity in the reactor prior to purging this radioactivity to the environment?

Response:

The radiation monitors located in the purge exhaust and vent stack would be used to ensure that the amount of radioactivity in the effluent is within the acceptable limits given in the technical specification limits. If the amount of radioactivity in the effluent is above the technical specification limits then the purge exhaust would automatically be returned to the reactor building.

18. In the event of an incident at Unit 2, how many workers would be available at any one time to deal with the emergency-at a time when the workers have been reduced in the first year and then in the second year and thereafter.

Is it possible or likely that workers from Unit 1 would be drawn to Unit 2 to help deal with an emergency?

Response:

According to the licensee (Letter from F.R. Standerfer, Director, TMI-2, GPU Nuclear Corporation to W.D. Travers, Director, TMI-2 Cleanup Project Directorate, NRC, November 5, 1987. Subject: Post-Defueling Monitored Storage Environmental Evaluation, NRC Comment Response) the level of direct employment for the PDMS program would be about 100 to 125 personnel during the transition year following the completion of current defueling activities and about 70-75 personnel thereafter until final cleanup. These workers would be available to deal with an emergency, although the number on site at any one time may vary and is unknown to the NRC.

Currently, fire, security, and medical emergency personnel are shared with Unit 1.

19. Does GPU Nuclear need an amendment to its license before PDMS is enacted.

Response:

Yes. Prior to entering PDMS, an amendment to the TMI-2 operating license would be required to align the technical specifications to plant conditions expected during long-term storage.

20. Since Unit 2 is in the 100 year flood plain, how will this affect its License prior to seeking approval for PDMS

Response:

As page 4.10 (Section 4.1.3) of Draft Supplement 3 indicates, the island on which both the TMI-1 and TMI-2 reactors are located is not within the 100-year flood plain; however, it is within the 500-year flood plain (0.2-percent chance of flooding in any given year) as determined by the U.S. Army Corps of Engineers. (See also Supplement 2, page A.8 and A.9). This will not affect the licensee's ability to seek approval for PDMS for two reasons. First, the island is diked for flood protection, and the dikes are inspected and maintained by the licensee. Second, TMI-2 flood procedures require that flood door panels be installed when the river elevation reaches 302 feet (92 meters). Installation of flood door panels effectively precludes the entry of river water.

21. Explain why the estimated occupational doses are so much higher for immediate clean-up.

Response:

The occupational dose range that was estimated for the alternative of immediate cleanup (300 to 3100 person-rem) is higher than the occupational dose range that was estimated for delayed cleanup (48 to 1500 person-rem) for the following reasons:

1) The 20 year period of post-defueling monitored storage would result in the decay of the principal radionuclides to levels approximately two-thirds the level that would be present during immediate cleanup.

2) It was assumed that robotics, decontamination and waste treatment technologies would allow further reduction in occupational dose levels during cleanup following PDMS.

22. Explain the subtle difference between the no-action alternative and the Licensee's proposal. What guarantees or laws will preclude the Licensee's PDMS proposal from becoming the no-action alternative?

Response:

Section 3.1.5 of Draft Supplement 3 describes the no-action alternative. Section 3.2 of Draft Supplement 3 describes the licensee's proposal. Section 3.1.5 states that the no-action alternative would be essentially the same as that described by the licensee's PDMS proposal except that neither preparations for PDMS nor subsequent actions to finish the cleanup would occur.

The NRC will not allow the licensee to place the facility into monitored storage until the necessary requirements for long term storage are met. The NRC regulations require that the license holders at nuclear power facilities take certain steps to assure that the facility will ultimately be decommissioned and equipment, structures, and portions of the facility and site containing radioactive contaminants are removed or decontaminated to levels acceptable for unrestricted use of the property.

23. Into what areas and how much money will the Licensee or the NRC put into research to develop technology for clean-up following PDMS? Will the NRC obtain a commitment from the Licensee to finance such development?

Response:

The NRC has no plans to develop technology for cleanup following PDMS. This task would be left to the licensee. No commitment will be obtained by the NRC from the licensee to finance further development of technology.

24. Will all of the waste generated since the on-set of clean-up and up to the placement of the plant in PDMS be removed from the island before the Unit is placed in PDMS?

Response:

For the evaluation in Draft Supplement 3, it was assumed that all of the waste associated with decontamination activities since the time of the accident would be removed from the island before Unit-2 is placed in PDMS. As discussed in Draft Supplement 3, some fuel debris would remain in the reactor vessel (page v), and some outside of the reactor vessel (page 2.18, section 2.1.3) and radioactive material will remain in many areas of the reactor building (Section 2.1.2) and some areas of the auxiliary and fuel-handling buildings (Section 2.1.3). The licensee in the document, "TMI-2 Cleanup Program Post-defueling Monitored Storage - Technical Plan" (TPO/TMI-188, January 1987) indicated that before the start of PDMS, "Radioactive material will have been removed or contained..."

TMI Alert

1. 2.1 The staff noted that, "The primary difference between an undamaged reactor at the end of its useful life and the licensee's PDMS proposal is that during PDMS relatively high levels of contamination would remain in the reactor building basement and a small amount of residual fuel would remain in the reactor coolant system [during] storage."

   What factual data are these conclusions derived from? How many "undamaged reactors at the end" of their "useful" lives have the NRC dealt with? Were technical experts from these plants consulted? If so, is their input a matter of public record? What other differences exist between these plants and GPU's PDMS plan? Was embrittlement a factor at these plants? What was the staffing levels at these plants?

   Response:

   The original PEIS (page 2.3; Section 2.1) indicated that "For full cleanup, all cleanup operations would be carried through to the point that the facilities were ready to initiate decommissioning or refurbishment operations." This is the condition of a undamaged reactor at the end of its useful life, at which time after the fuel has been removed, it is ready for decommissioning or refurbishment operations. The statement cited in the above question was meant as a comparative statement rather than a quantitative statement. The comparison between an undamaged reactor at the end of its useful life, and the licensee's PDMS proposal for the TMI-2 reactor undamaged reactors was made to indicate that unlike TMI-2, undamaged reactors have not had large quantities of radioactive water dumped into their basements and have not had fuel debris dispersed through their reactor coolant system. No comparison of potential for embrittlement or of staffing levels was implied.

   The NRC has had considerable experience with reactors that have not had a significant accident before the end of their useful lives. Examples include Humboldt Bay, Dresden 1, Indian Point 1, LaCrosse, Shippingport, Elk River, and Carolina-Virginia Tube reactor. These reactors differed from each other and from Three Mile Island, Unit 2, in design, operating history, and power levels.

2. 2.1.1 The staff argued that, "The reactor containment building is uniquely designed and constructed to maintain its structural integrity (with almost no leakage) during a wide variety of accidents."

   How long after an accident was the RCB designed to maintain its integrity? Was it specifically designed to house radioactive waste materials for an indefinite period of time? If not, would not storage of such wastes necessitate a license amendment?

Response:

The reactor containment building was designed to maintain its integrity during a peak accident pressure of 60 psig allowing only 0.2 percent leakage during the first 24 hours following the accident, and 0.1 percent leakage per day thereafter. However, the accident that occurred at TMI-2 was not an accident of this proportion, having reached a peak accident pressure of 28 psig. The reactor containment building was designed to maintain its integrity for a period of 40 years, whether or not a design base accident occurs. Construction did not begin until late 1969.

The reactor building was not designed specifically to house radioactive waste for an indefinite period of time. The current NRC regulations do not allow for an indefinite storage of the facility.

3. 2.4 How permanent are "permanent dose reduction techniques?"

Response:

The term "permanent dose reduction techniques" refers to methods that permanently remove the source of radiation from the area where it is located. This term was used to distinguish these methods from the dose reduction technique of shielding the source of radiation by placing structures on or around it to attenuate the dose rate.

4. 2.1.1 "Sectioning and disposal of the reactor internals and reactor vessel are not considered part of the cleanup because radiation levels expected from these components would be no higher than in a normal reactor nearing the end of its life."

What are "sectioning and positioning of the reactor internals" part of? What if radiation levels are incorrect? What exactly are the radiation levels of a "normal reactor at the end of its life?" What constitutes a normal reactor?

Response:

Sectioning and disposal of the reactor internals and reactor vessel are considered part of the next phase in the life of a reactor, the decommissioning or recommissioning process, because this activity would also occur during decommissioning or recommissioning of a reactor facility that has not undergone a significant accident. In other words, this is not an action that is necessitated in order to clean up the facility as a result of the accident.

Even if radiation levels in the reactor internals and reactor vessel are found to be higher than expected during sectioning and disposal operations, little or no impact would be expected, because additional shielding or distance could be used to reduce occupational dose. However, because of the short length of time the TMI-2 reactor operated (less than 14 months), the quantity of the activation products in the reactor internals and in the reactor vessel are less than the quantity that is present in a reactor that has operated for longer than 14 months and much less than the quantity that would be present in a reactor that had operated for 40 years. It is not likely that this assumption is incorrect. Measurements taken on the lower grid rib section and plenum confirm that radiation levels are no greater than expected on reactor internal components.

The radiation levels emitted from the reactor internals and reactor vessel will vary among nuclear reactor facilities, depending on the material used to construct the vessel and internals, the operating history, and the operating power. The statement in question was meant as a qualitative statement made for the purpose of comparison to explain why certain activities were considered to be part of the decommissioning or recommissioning process rather than part of cleanup. We do not expect to compare the absolute radiation levels in the reactor internals and reactor vessel of the TMI-2 reactor with the levels in normal reactors at the end of their useful lives.

The term "normal reactor" as used in this supplement refers to a reactor that has not undergone a significant accident. This term will be included in the nomenclature section of the final report.

5. 2.1.4 What unique problems will the AFHB pose since it "was not designed to be leak free..." during a "...variety of accidents?" How much, and just exactly what, leaks from the AFHB? What are the dose levels found in AFHB at the end of its life?"

Response:

Buildings in general are not designed to be leak free, especially under accident conditions. Because the dose levels expected to be present in the AFHB at the end of the current defueling efforts are expected to be similar to those found in the AFHBs of operating reactors, no unique problems would be posed by the TMI-2 AFHB.

The general area dose levels in the AFHB are below 2.5 mR/hr in many areas although they do approach 15 mR/hr in some of the cubicles that contain equipment. This is similar to dose levels that would be found in the AFHBs of operating plants after operation for 40 years.

6. 2.2.1 Why weren't new calculations taken concerning the number and quantity of remaining radionuclides? Does the NRC or GPU have a comprehensive inventory of the radionuclides released since the accident? Is it possible for radiation levels to shift or relocate from one section of the plant to another? If so, isn't [it] possible that sections designated to have certain radiation levels may now be inconsistent with GPU's endpoint criteria?

Response:

New calculations were made to determine the number and quantity of radionuclides expected to be in the facility following the current defueling efforts. The results are shown in Table 2.4 and the assumptions that were made in support of the calculations are given in sections 2.2.1 through 2.2.3. These calculations were based on the amount of radioactive material present at the time of the accident as given by the ORIGEN-2 computer code.

GPU measures the radioactive material releases and reports them to the NRC per the requirements of Section 5.6.1.C of Appendix B to the Recovery Technical Specifications. The releases are reported quarterly for gaseous effluent releases, liquid releases, and solid waste and irradiated fuel shipments. However, due to the nature of the accident and the method by which the material has been removed from the reactor and shipped offsite, we can not provide a comprehensive inventory of every radionuclide since the time of the accident.

The following methods will shift or relocate radiation levels from one section of the plant to another;

1) Movement of radioactive material by personnel, either advertently or inadvertently,
2) Movement of radioactive material by animals or insects,
3) Movement of radioactive materials by water or air transport.

The licensee makes at least monthly measurements of the amount of radioactivity present in the TMI-2 facility. These measurements are used to determine the decontamination progress that has taken place to date, and can be used to identify any relocation of radiation levels from one section of the plant to another. These measurements are also used to ascertain whether the endpoint criteria have been met.

Questions from the Transcript

Kenneth Miller
Director, Health Physics Division
Hershey Medical Center

p. 38 I would like to know, can you do a comparison for us between the contamination levels that will exist at the end of the defueling period with the levels you keep referring to that exist at the end of a useful life of an operating reactor? You're talking about a factor of 2, 10, 1000?

Your comment -- you kept referring to the fact that you would like to see the plant cleaned up to the point where it matched the levels of contamination present at the end of the useful life of an operating reactor.

Can you give us some sort of a comparison? Maybe they are already lower and you don't have to do anything -- but I doubt that.

Response:
Answered in transcript, pages 39 and 40.

p. 40 So the required additional cleanup will be strictly concentrated on those areas that are still unreasonably high?

Response:
Answered in transcript, pages 40 and 41.

p. 50 Are the funds currently available to do an immediate cleanup?

Response:
Answered in transcript, page 50.

Thomas Gerusky
Director, Bureau of Radiation Safety
Pennsylvania Department of Environmental Resources

p. 41 You made a comparison between a four year cleanup and a twenty-four year delayed cleanup with a total exposure comparison for four years and twenty-four years.

Would there be any environmental impact or any exposure to the public following the immediate four year cleanup and after the twenty-four cleanup that has not been taken into consideration in comparing the two?

Response:
Answered in transcript, pages 41 and 42.

p. 42 But you don't think that the public and we ought to have a feel for what those doses are, what are those environmental impacts are, for the same periods of time until decommissioning, assume you go to decommissioning?

Response:
Answered in transcript, page 42.

p. 42 You're comparing twenty-four years versus four years, and shouldn't you compare 24 to 24? That's what I'm asking. If you're going to an end point, shouldn't the end point be the same for the exposures for both options?

Response:
Answered in transcript, page 43.

p. 43 Is that in the document?

Response:
Answered in transcript, page 43.

p. 44 And that's for twenty years?

Response:
Answered in transcript, pages 44 and 45.

p. 50 Do you have an estimate of cost of decommissioning TMI 1?

Response:
The recently enacted decommissioning rule requires around $100 million to be set aside to assure adequate funds for decommissioning.

p. 51 Is that [funds currently available to do an immediate cleanup] out of the cleanup fund or out of additional funds that the utility would have to spend on its own.

Response:
Answered in transcript, page 51.

Arthur Morris, Chairman
Mayor of Lancaster, Pennsylvania

p. 45 Was there any attempt done to analyze, or is it part of the study, to analyze the ability of the licensee to finance this cleanup? Whether they could financially afford to do it immediately or whether, in fact, in twenty years from now whether they'll be able to finance or be afford to do it at the time?

Response:
Answered in transcript, page 45.

p. 104 What kind of financial responsibility the NRC would hold them to?

Response:
The NRC will not require the licensee to set aside funds exclusively for the final cleanup of TMI-2, however, the recently enacted decommissioning rule requires around $100 million to be set aside to ensure adequate funds for decommissioning.

Joel Roth
Advisory Panel Member, Representing the Public

p. 47 Is there any provision that the NRC can make to guarantee that the funds be available at that time?

Response:
Answered in transcript, page 47.

Francis Skolnick, Susquehanna Valley Alliance

p. 54 The NRC speaks of a twenty year storage period but provides no rationale for choosing this number.

Response:
Because no information was provided by the licensee as to the length of the storage period, a storage period of 20 years was assumed because this will approximately coincide with the end of TMI-2's operating license in the year 2009.

p. 54 The NRC tells us that immediate cleanup would require additional emergency allocations. That's in EIS Page 2.33.

Response:

Page 2.33 of draft Supplement 3 to the PEIS states, "Immediate cleanup without PDMS could require additional emergency allocations". It has not yet been determined whether or not it will require additional emergency allocations for disposal of waste.

p. 55 I ask why generate more water when we have already accumulated the major medium for decontamination?

Response:

If the accident-generated water is available for use at the time of the final stage of cleanup, no action would preclude its use during decontamination. If it is not available, an additional source of water would be required. Because we wished to address the impact of storage and final cleanup (the impact of disposal of the accident-generated water was addressed in Supplement 2) we assumed that an additional source of water would be used.

p. 56 Table 2.4 in the EIS, which shows an estimate of the maximum amount of radionuclides left and their location... We want to know upon what information this table might be based. Furthermore, we want to have a complete accounting of the radionuclides present in the core at the time of the accident.

Response:

The information on which the list of radionuclides and their quantities (as given in Table 2.3) is based, is discussed in Section 2.2 of Draft Supplement 3. That information is reiterated as follows:

1) The inventory of radionuclides that were estimated to be present at the time of the accident was obtained from two separate analyses by two separate groups (GPU and the Electric Power Research Institute) using the ORIGEN-2 computer code.

2) The effect that radioactive decay would have had on the inventory of radionuclides between the time of the accident and the projected completion of defueling was included. The results are shown in Table 2.3 of Draft Supplement 3, which contains a list of the radionuclides that would have inventories of greater than 1 curie on January 1, 1989.

3) The effect cleanup has had on the radionuclide inventory was considered. For instance, a large quantity of the radionuclides have been removed during the defueling process. Krypton-85, a gas, has been vented to the atmosphere. Large quantities of cesium, strontium, and other water soluble radionuclides have been filtered out of the accident-generated water and disposed of along with the ion-exchange resins in the low-level waste site.

4) The assumptions regarding the location of the radionuclides are based on the chemical and physical form of the radionuclides as discussed in Sections 2.2.1, 2.2.2, and 2.2.3.

Due to the nature of the accident and the method by which the material has been removed from the reactor and shipped offsite, we can not provide a complete accounting of every radionuclide since the time of the accident.

p. 56 Looking at just two of the radionuclides, tritium, which the NRC failed to mention was an important activation product, and Krypton 85, it is impossible to account for all of both of these radionuclides.

There were over 8,800 curies of tritium and over 97,000 curies of Krypton in the reactor at the time of the accident. How does the NRC end up with less than 1 curie of both tritium and Krypton 85?

Response:

Sections 2.2.2.1 and 2.2.2.2 of Draft Supplement 3 to the PEIS discuss the assumptions used to estimate the amount of krypton-85 and tritium (respectively) in the reactor after the current defueling process is complete.

Krypton-85, a gas, was released to the reactor building during the accident and was subsequently vented to the atmosphere. Although some krypton-85 may have remained trapped between and in fuel material in the reactor vessel, during the defueling process this fraction of the krypton-85 was either released into the reactor building and removed through the stack or remained with the fuel material and shipped in the canisters to Idaho. The radiation monitor in the stack has been used to measure all the effluents from the reactor building. During the past several years, no krypton-85 has been measured, indicating that it is no longer present in measureable quantities in the reactor building.

Tritium was produced within the reactor fuel by several mechanisms including activation and ternary fission. Greater than 90 percent of the tritium produced in a pressurized water reactor such as TMI-2 is produced by ternary fission. Because tritium has the same physical properties as water, once the water is removed from the facility (one of the assumptions in this evaluation was that all the water would be removed from the facility) the tritium has also been removed. It was further assumed that any tritium remaining as dampness in the facility would either exchange with the hydrogen in the air or evaporate during the first few months after removal of the water.

p. 57 The approval of TMI to become a site for the storage of radioactive waste raises questions about regulatory procedures and, furthermore, the acceptability of this plan to the State of Pennsylvania.

If cleanup were to continue presently, then the waste would go to out of the state sites. If it is delayed, it will largely remain within the state.

I ask how can the NRC dismiss the question of the impact of the waste disposal by saying that it would be the subject of an analysis elsewhere? The disposal of waste at TMI is a major issue to be dealt with at this time and it is in keeping with the requirements of the National Environmental Policy Act.

Response:

The environmental impact of waste disposal at a commercial low level waste disposal site, is the subject of an environmental evaluation specific to the chosen site, which must be completed before the site can be licensed. Waste streams outside those evaluated during the environmental evaluation for the site will not be allowed for burial. The environmental evaluation for a regional burial site must be specific to the environmental characteristics of the site, and must also address all types of wastes that will be accepted into it, including wastes from hospitals and university research laboratories. Wastes from TMI-2 will not be accepted at a regional site, until the site is licensed.

p. 58 How will the NRC deal with the fact that Unit 2 is in the hundred year flood plain? Will it have to maneuver the regulations in some way that TMI will be exempt from the requirements? Will TMI be able to satisfy the ground water intrusion criteria?

Response:

Page 4.10 (Section 4.1.3) of Draft Supplement 3 indicates, the island on which both the TMI-1 and TMI-2 reactors are located is not within the 100-year flood plain, however, it is within the 500-year flood plain (0.2-percent chance of flooding in any given year) as determined by the U.S. Army Corps of Enigineers (see Supplement 2, page A.8 and A.9).

The regulations will not be altered to exempt TMI from requirements.

As indicated on page 3.10, of the PEIS Draft Supplement 3, quarterly ground water monitoring would be continued during PDMS to ensure that little or no out-leakage occurs from plant buildings.

p. 65 I suppose then another question which I would have to ask is if cleanup is delayed and resumed in whatever period of time, whenever, and they need -- I think its over a million gallons for clean up -- would that water be accident generated water?

Response:

The definition of accident-generated water is presented in the nomenclature list of Draft Supplement 3 as follows:

> On Fegruary 27, 1980, an agreement executed among the City of Lancaster, Pennsylvania, Metropolitan Edison Company and the NRC defined "accident-generated water" as:
>
> - Water that existed in the TMI-2 auxiliary, fuel handling, and containment buildings including the primary system as of October 16, 1979, with the exception of water which as a result of decontamination operations becomes commingled with nonaccident-generated water such that the commingled water has a tritium content of 0.025 uCi/mL or less before processing.
>
> - Water that has a total activity of greater than 1 uCi/mL prior to processing except where such water is originally nonaccident water and becomes contaminated by use in cleanup.
>
> - Water that contains greater than 0.025 uCi/mL of tritium before processing.

The water generated during final cleanup would not meet the first two definitions of accident-generated water, and could meet the third definition if the quanitity of tritium in the water is greater than 0.025 uCi/ml. This translates to 23.7 curies of tritium in the entire 1,000,000 gallons used during cleanup. Because less than 1 curie of tritium is expected to remain in the facility after defueling and drainage of all liquids, the water generated during cleanup would not meet the legal definition of accident-generated water.

Eric Epstein, Three Mile Island Alert

p. 75 We would appreciate it if GPU or the NRC could furnish a complete inventory of where all the radioactive materials have gone since the accident.

Response:

Due to the nature of the accident and the method by which the material has been removed from the reactor and shipped offsite, we can not provide a complete inventory by isotope of where all the racioactive materials have gone since the accident.

p. 75 In the document is ventilating the reactor building before each entry the same as purging it?

Response:

Yes.

p. 75 How will the liquid releases to the Susquehanna River following PDMS differ in composition to the 2.3 million gallons of radioactive water currently stored at TMI.

Response:

The liquid releases to the Susquehanna River following PDMS would be recycled through ion exchange columns as necessary, to ensure that the release rates to the Susquehanna River are below technical specificiation limits. The liquid releases would be similar in composition to the accident-generated water after processing through ion-exchange systems, except that the liquid releases following PDMS would contain only trace amounts of tritium. Furthermore, some of the shorter half-life isotopes (such as manganese-54, cerium-144 and praseodymium-144) would have decayed to negligible levels.

p. 75 Also, just as a question, and I think I know the answer, is the public entitled to intervene if the indefinite storage option is implemented?

Response:

At the time that the licensee requests an amendment to the TMI-2 operating license allowing PDMS, the public will be given the opportunity to request a hearing.

p. 75 The final question, and I think I know the answer to this also. If the cost of the cleanup is figured in 1988 dollars, then estimates for delayed cleanup are imprecise and inaccurate.

Response:

All costs are figured in 1988 dollars as indicated and discussed on page 3.29 (Section 3.2.6), page 3.42 (Section 3.3.6) and p 5.4 (Section 5.1). The cost estimates in the PEIS are given as ranges for the purpose of comparison only. These numbers represent the best estimate of cost at the time the supplement was prepared.

p. 75 What I was curious is if the NRC factored into the economic costs the costs for retraining and rehiring workers that have been gone for some twenty years.

Response:

The additional cost from retraining workers was addressed on page 3.29 (section 3.2.6). It was indirectly factored into the cost estimates by assuming that immediate cleanup would require 3 to 4 years and cleanup following PDMS would require 4 years for completion.

Vera Stuchinski, Chairperson, Three Mile Island Alert

p. 80 What's to stop GPU from making their own rules?

Response:

The NRC will have a continuing onsite presence and will require the licensee to maintain the facility in accordance with all applicable rules and regulations.

p. 82/83 Now, I'd like to ask Dr. Travers why the staff does not consider PDMS in the same manner as storage of the tritiated water. If a low-level waste site license would be required for storage of the water, why isn't it required for PDMS?

Response:

Answered in transcript, pages 83 to 85.

p. 86 Do you really feel that that there would be significant decay of the radioactive material within twenty years of any long-lived radionuclides in the reactor?

Response:

Answered in transcript, page 86.

Kay Pickering, Office Coordinator, TMIA

No questions

Ed Trunk, Professor of Mechanical Engineering, Pennsylvania State Univ.

p. 90 The question was why are we considering this question when we had a timetable before us and we're going down that timetable. Why are we considering this? Why is there a change in the timetable before us right now?

Response:
Answered in transcript on page 90.

Joyce Corradi, Director, Concerned Mothers and Women

p. 92 My first question is in reference to what was told to me tonight. In the presentation by the NRC, they said that in twenty years there would be three million or more people in the area that they were relating to for their dose rate.

I'd like to know where they got their projection and how they got that projection.

Response:
Answered in transcript on page 92.

p. 92 I'd like to know from Mr. Standerfer where he got it from and how it was calculated.

Response:
Answered in transcript by Frank Standerfer, GPU, on pages 95 and 96.

p. 96 If, indeed, this is a criteria by which they were using to get dose rates, I should like to know where they came from, the year point end of them, and how valid and updated they are.

Response:
Frank Standerfer, GPU, will supply by response by next meeting.

Debra Davenport, Member, Concerned Mothers and Women

p. 93 I want to know what the licensee plans to do to deal with the materials that are directly under the reactor vessel. Is this included in any of the assessments of removal of materials from the plant?

Response:

Answered in transcript by Frank Standerfer, GPU, pages 97 and 98.

p. 94 But what is under the reactor vessel? What is passed -- I know something in the book with the nozzles going into the vessel, but what about the tubes leading into the nozzles. What fuel is in there?

So I really question whether we're being told about all the fuel that's in the plant and whether there is a full assessment made on removing those fuels.

Also, I really wonder why, over a long period of time, we repeatedly seem to have a drawback from explaining to the public what might be under the reactor vessel in the basement.

Response:

Answered in transcript by Frank Standerfer, GPU, pages 97 and 98.

p. 97 I want to know, are they going to check that area under the reactor vessel, because this has been an off-again and on-again thing for the past year. Are they going to say what's there?

Response:

Answered in transcript by Frank Standerfer, GPU, page 98.

p. 98 Is it going to be left there and how much of it is there?

The second one, in the inner core detector tube, is materials from the -- or any materials going under the reactor vessel. When are we going to know about this?

Response:

Answered in transcript by Frank Standerfer, GPU, page 98.

p. 98 Why is it to radioactive down there and you can't get in?

Response:

Answered in transcript by Frank Standerfer, GPU, page 98.

p. 98 Why wouldn't it be the same -- as the rest of the --

Response:

Answered in transcript by Frank Standerfer, GPU, page 99.