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February 6, 1981
LL2-81-0034

TMI Program Office
Attn: Mr. Bernard J. Snyder
Program Director-TMI Office
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Sir:

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

Your letter, dated December 24, 1980, requests that we submit to you a comprehensive written safety evaluation of the SDS. In our letter, LL2-81-0013, dated January 19, 1981, we committed to supply the requested information by February 6, 1981. This letter fulfills that commitment.

The enclosure to this letter documents our safety evaluation. As a result of the performance of this safety evaluation we conclude that operation of the Submerged Demineralizer System neither presents an unreviewed safety question nor requires a change to the technical specifications.

It is our intent that the enclosed evaluation meets your requirements. Should you wish to discuss this matter, please contact Mr. L. J. Lehman, Jr. (717) 948-8599 of my staff.

Sincerely,

G. K. Hovey
Vice-President and
Director, TMI-2

GKH:LJL:djb

Enclosure

cc: L. H. Barrett, Deputy Program Director

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SUBMERGED DEMINERALIZER SYSTEM

SAFETY EVALUATION

BACKGROUND

The present mode of operation of TMI-2 is governed, in part, by the Interim Recovery Technical Specifications, promulgated by the Nuclear Regulatory Commission order dated February 11, 1980. These Tech Specs do not relieve the licensee of compliance with the rules and regulations that apply to domestic production and utilization facilities (10CFR 50). 10CFR 50.59(a)(1) states:

"The holder of a license authorizing operation of a production or utilization facility may (i) make changes in the facility as described in the safety analysis report, (ii) make changes in the procedures as described in the safety analysis report, and (iii) conduct tests or experiments not described in the safety analysis report, without prior Commission approval, unless the proposed change, test, or experiment involves a change in the technical specifications incorporated in the license or an unreviewed safety question."

PURPOSE

The purpose of this safety evaluation is to provide a documented basis for the following conclusions:

- (1) Operation of the SDS does not require a change to the TMI-2 technical specifications.
- (2) Operation of the SDS is not an unreviewed safety question.

EVALUATION CRITERIA

The evaluation criteria to be used for the determination of an unreviewed safety question are specified in 10CFR 50.59(a)(2) which states:

"A proposed change, test, or experiment shall be deemed to involve an unreviewed safety question (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced."

The evaluation criterion for the determination of the requirement to change the technical specifications is based upon the intended operations of SDS and the impact on existing Interim Recovery Technical Specifications.

SAFETY EVALUATION

- (1) Evaluation of SDS operation against 10CFR 50.59(a)(1).
 - (a) Implementation of SDS does involve a change in the facility as described in the SAR, even though the change is only temporary in nature to be used specifically for TMI-2 recovery.
 - (b) Implementation of SDS does involve a change in the procedures as described in the SAR; the procedure of the processing of radioactivity contaminated waste is addressed. However, because the SDS employs a different methodology for radioactive waste processing than is described in the SAR, it is considered that this specific procedure for waste processing is not addressed.

- (c) The operation of SDS is intended for the processing of high-level radioactive waste waters; it is not intended to be a test or experiment.

Because operation of the SDS is considered to be a change in the facility and procedures as described in the SAR, it is necessary to evaluate SDS operations against the criteria of 10CFR 50.59(a)(2).

- (2) Evaluation of SDS operation against the criteria of 10CFR 50.59(a)(2).

Addressing each of these criteria in turn is presented below.

- (a) The Probability of Occurrence of an Accident or Malfunction of Equipment Important to Safety Previously Evaluated in the SAR may be Increased.

The SDS flowpath will provide for radionuclide removal of the process flow stream. From the containment sump the water will be pumped via the prefilter and final filter, to four 15,000 gal. (ea.) tanks, referred to as the tank farm. The tank farm tanks operate as one tank, they are interconnected with valve-less welding piping. The feed pump suction well from which SDS influent water is supplied, is located at the same elevation as tank farm tanks. The water level will rise in the well as the tanks are filled. The suction well is equipped with level indication that is alarmed on high level. Should the water level continue to rise, a backup level device will be actuated to automatically close the fill valve to the tank farm and preclude overflow of the suction well with containment sump water.

Contaminated water is transferred from the suction well, via the SDS feed pump through welded stainless steel piping, to the SDS ion exchange vessels through quick disconnect couplings. This quick disconnects and ion exchange vessels are contained in a leakage containment box which contains spent fuel

pool water. Any leakage from the quick disconnects which occurs during routine operation of SDS or when connecting or disconnecting ion exchange vessels will be contained within the containment boxes, diluted by pool water, and treated by the leakage ion-exchange system prior to return to the spent fuel pool.

SDS processing is performed by flowing water through three stainless-steel zeolite containers in series for Cesium and Strontium removal, one additional stainless-steel ion-exchange vessel specifically loaded with resin materials for Strontium removal, and into the EPICOR II system for final SDS effluent polishing for removal of remaining trace radionuclides, such as Antimony, and recalcitrant species of Cesium and Strontium. EPICOR-II operation has been authorized by order of the Commission dated October 16, 1979. Processed water will be stored in the Processed Water Storage Tanks on Three Mile Island. No liquid effluents resulting from SDS operation are planned to be released to the environment at this time.

Operation of SDS will be performed under strict administrative procedural control. Operator training is on-going with operator walk-through of the operating procedures to be performed during pre-operational testing. These walk-throughs will provide the opportunity for "hands-on" experience by operations personnel to gain system familiarity as well as to actually test the operating procedures to be used prior to actual processing of contaminated water. Furthermore, the procedures to be used for operation of the SDS will be submitted to the Nuclear Regulatory Commission for review and approval prior to use in accordance with Technical Specification 6.8.1.

(b) The Consequences of an Accident or Malfunction of Equipment.

The Technical Evaluation Report (TER) submitted to the NRC on April 10, 1980 (TLL 160), concerning the SDS contains (in chapter 7) several hypothetical accidents. The accidents presented, though highly unlikely and improbable,

present bounding conditions for accident scenarios. At the time of generation of the afore-mentioned document, the source terms used were representative of contamination levels of the sump water. Because of the interval of time that has passed since development of the TER accident analysis, source terms are approximately one-third the value reported in the TER due to radionuclide decay. Therefore, because of the lower source terms, the TER conclusions remain valid. Detailed information is provided below.

Inadvertant pumping of containment sump water into the spent fuel pool.

The scenario for this hypothetical accident remains the same.

Occupational Exposure Effects:

Because of the reduced source term, the calculated maximum exposure rate at six feet above the pool surface is reduced to approximately 115 mr/hr. Conclusions regarding the occupational exposure effects of this hypothetical accident remain the same as the TER conclusions except for the reduction of the dose rate.

Off-site Effects:

Radiological effects of this hypothetical accident are assumed to result from two contributing factors. They are:

- o Direct radiation exposure.
- o Airborne contamination.

Direct radiation exposure at the site boundary is calculated to be 4.5×10^{-7} mr/hr. This calculation is based on the following assumptions:

- o The isotopes of concern are Cs-134 and Cs-137.
- o The fuel pool can be considered as a point source for site boundary direct dose calculations.
- o No source self-absorption occurs.
- o The fuel pool wall and the fuel handling building wall provide 3' of concrete shielding.

- o The pool leakage cleanup ion-exchanger system will remove activity from the spent fuel pool. This system will process the pool water at the rate of approximately 100 gpm.

Airborne contamination may be generated as a result of direct evaporation from the pool surface to the Fuel Handling Building atmosphere. The path to the unrestricted environment requires that the airborne radionuclides pass through the plant HEPA filters prior to discharge via the plant vent. Analysis of this hypothetical occurrence is based upon the following assumptions:

- o Activity spilled into the pool is uniformly distributed.
- o The pool leakage cleanup ion-exchanger system will remove activity from the spent fuel pool. This system will process the pool water at the rate of 100 gpm.
- o The isotopic inventory of the spent fuel pool is conservatively assumed to remain constant for a period of one week.
- o The spent fuel pool volume is 233,00 gallons.
- o The evaporation entrainment factor is conservatively estimated to be 10^{-6} .
- o Plant ventilation system HEPA DF is 10^2 .
- o Air flow across the surface of the spent fuel pool is 5500 cfm.

Based on the above-specified assumptions, airborne contamination released to the atmosphere as a result of this hypothetical accident is approximately $4 \frac{\mu\text{Ci}}{\text{wk}}$ of the Cs-137 isotope, approximately 3.75% of the normal operation atmospheric release of this isotope. This percentage increase is valid for other total body dose contributing isotopes. Normal operation of SDS results in an estimated total body exposure of approximately 3.6×10^{-3} mrem/yr. from all isotopes to the maximally exposed individual. The increase in total body exposure revises the estimated total body exposure to 3.735×10^{-3} mrem/yr. This increased exposure is 0.0747% of the allowable dose exposure of 10CFR 50, Appendix I of 5 mrem.

Pipe rupture on filter inlet line (above water level).

The scenario for this hypothetical accident remains the same.

Occupations Exposure Effects:

Because of the reduced source term, the significant effects identified in the TER are as follows:

1. The maximum exposure rate at the surface of the contaminated floor area is estimated to be approximately 3.6 Rem/hr.
2. The maximum beta exposure rate at a point three feet above the surface of the contaminated floor area is estimated to be 128 Rad/hr.

Conclusions regarding occupational exposure effects of this hypothetical accident are the same as the TER.

The estimated occupational exposure effects are based on the following assumptions:

- o Contaminated water sprays into the air from behind the lead shielding. Approximately 675 gallons of sump water is released directly into the spent fuel pool and 75 gallons spreads over a surface area of 200 ft².
- o Primary contributors to the estimated dose rate are Sr-89, Sr-90, and Cs-134, Cs-137.

Off-site Effects:

Off-site radiological effects from this hypothetical accident are assumed to result from two contributing factors. They are:

- o Direct radiation exposure.
- o Airborne contamination

These estimated effects are based on the following assumptions:

- o The isotopes of concern are Cs-134 and Cs-137.
- o The distance to the closest off-site point is approximately 200 meters.
- o The spent fuel pool can be considered a point source for exposure estimates at a distance of 200 meters.

- o There is no significant source self-absorption.
- o The fuel pool wall and the Fuel Handling Building wall provide a direct dose attenuation equivalent to three feet of concrete.

Direct radiation exposure estimates indicate that radiation exposure at the site boundary will increase by approximately 6.75×10^{-7} mRem/hr.

Airborne contamination may be generated as a result of this hypothetical accident. The assumptions used to estimate these consequences are the same as those used for the airborne contamination estimates of the previous hypothetical accident. Based on these assumptions, airborne contamination released to the atmosphere as a result of this hypothetical accident is approximately 6.3 μ Ci/wk of the Cs-137 isotope, approximately 5.63% of the normal operation atmospheric release of this isotope. The percentage increase is valid for other total body dose contributing isotopes. Therefore, the increase in total body exposure resulting from this hypothetical accident is approximately 0.20^3 mRem/yr. The total body exposure, including the effects of this postulated accident, is approximately 3.8×10^{-3} mRem/yr, approximately 0.076% of 10CFR 50 Appendix I limits of 5 mRem for normal operations.

Inadvertant lifting of prefilter above pool surface.

The scenario specified in the TER remains the same. The analysis has been performed based on the following assumptions:

- o A failure in the crane control system results in the "dragging" of the filter over the edge of the spent fuel pool.
- o The prefilter is loaded with 100 Curies of 8-emitters.
- o The minimum distance for exposure calculations is 4.57 meters. The prefilter can be considered to be a point source.
- o There is no source self-absorption.
- o There is no container shielding.
- o There is no environmental release as a result of this hypothetical accident.

Occupational Exposure Effects:

The calculated exposure rate at a distance of 15 feet from prefilter in air is 21 R/hr. The effects identified in the TER are valid.

(c) The Possible Creation of a Different Type of Accident or Malfunction.

Additional accident postulations are given below.

- (1.) Possible rupture of zeolite ion exchange vessel in storage and release of contaminated zeolite resins to the spent fuel pool.

In the unlikely event of this improbable occurrence, environmental consequences of no significance will occur. Even though the entire contents of the ion-exchange vessel may be released to the spent fuel pool the contaminated zeolite resins will fall to the bottom of the pool. Radionuclides contained within the zeolite, primarily the Cesium isotope, will not be released to the pool water (and hence to the environment) in significant quantities; they will remain adsorbed onto the zeolite resins. A significant radiological hazard may exist for cleanup of the resins from the pool floor. However, because a significant hazard will not be presented as a result of this occurrence, due to pool shielding, sufficient time exists to develop adequate cleanup procedures and/or cleanup equipment. Furthermore, rupture of a zeolite ion-exchange vessel in the spent fuel pool is highly unlikely. Two potential mechanisms for vessel rupture have been identified: (1) container corrosion, and (2) drop of vessel in the pool. Vessel rupture, as a result of corrosion effects, is regarded as an occurrence of such low probability to be incredible. Zeolite resins are not known to cause a pH

change in residual water; the ion-exchange vessels are fabricated from stainless steel. Not only is a corrosion-causing mechanism absent, the vessel material is extremely corrosion resistant. Assuming that a vessel drop in the pool occurs, it is highly unlikely that the vessel will break open and allow its contents to spill on the fuel pool floor. In the extremely unlikely event that the vessel does break open and allow the contaminated zeolites to spill on the pool floor, significant quantities of radionuclides would not be released to cause danger to the public health and safety as a result of airborne particulate release. Cleanup of the spilled contaminated resins would be performed under strict administrative control. Cleanup procedures would be reviewed and approved by the Nuclear Regulatory Commission. Sufficient time would be available for procedure development and approval and personnel mobilization.

(2.) Drop of shipping cask loaded with spent zeolite vessel during transfer from the spent fuel pool to the truck bay.

Present processing plans do not require that transfers of vessels from the spent fuel pool to the truck bay filled with contaminated materials to be performed. At the completion of vessel radionuclide loading, that vessel will be removed from service and placed in a storage location in the spent fuel pool. Should processing plans be changed such that transfers as described above are required, analysis of this postulated accident will be performed.

(d) Reduction in Safety Margin Defined in Bases of Technical Specifications.

The focus of this criteria is on the margin of safety as defined in the bases for any technical specification. Since the radwaste system is not addressed in the technical specification bases, this consideration is not applicable.

Evaluation of Requirement to Amend the Present Recovery Technical Specifications.

Implementation of SDS operations for decontamination of the contaminated water presently in the containment building requires no change to the existing TMI-2 Interim Recovery Technical Specifications. Liquid effluents will not be released to the environment directly from SDS operations; SDS effluent will be placed in the Processed Water Storage Tanks.

Furthermore, gaseous effluents resulting from SDS operations will traverse existing gaseous effluent flow paths. We do not perceive the requirement to change the maximum permissible concentrations or the instrument configuration or setpoints specified in Appendix B of the Interim Recovery Technical Specifications.

Finally, as specified in the Technical Specifications, Article 3.9.14, we will not process and discharge the water in the Reactor Building sump and the Reactor Coolant System unless NRC approval is received.

CONCLUSION

The purpose of documenting this safety evaluation for the Submerged Demineralizer System is to provide the following conclusion: design, construction and operation of the SDS does not present an unreviewed safety question. This conclusion is supported by the below listed facts:

- (1.) The SDS does not present the opportunity to increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report.
- (2.) The SDS does not present the opportunity to create the possibility of an accident or malfunction of a different type than any evaluated previously in the safety analysis report.
- (3.) The SDS does not present the opportunity for reduction of the margin of safety as defined in the basis for any technical specification.

Processing water in the containment building will be performed in compliance with the existing TMI-2 Interim Recovery Technical Specifications. No license amendment in the form of a change to the Technical Specifications is required.