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U.S. NUCLEAR REGULATORY COMMISSION

TMI Program Office
Attn: Mr. L. H. Barrett
Deputy Program Director
US Nuclear Regulatory Commission
c/o Three Mile Island Nuclear Station
Middletown, PA 17057-0191

Dear Sir:

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

Attached for your review and approval is an addendum to the Underhead Characterization Safety Evaluation which covers the proposed Core Sampling Program. This document should be considered an Addendum to both versions of the SER's previously submitted for your approval.

This document demonstrates that the proposed core samples of the TMI-2 core can be accomplished without undue risk to the health and safety of the public.

It is GPUNC's intention to perform this activity following the completion of the Underhead Characterization Study and the Core Topography Program as presented to the NRC in previously submitted documents.

If you have any questions, please contact Mr. J. J. Byrne of my staff.

Sincerely,

B. K. Kanga

B. K. Kanga
Director, TMI-2

BKK/RBS/jep

Attachment

CC: Dr. B. J. Snyder, Program Director - TMI Program Office

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Attachment 1

TMI-2 Core Debris Grab Samples Safety Evaluation

1.0 Purpose and Scope

The purpose of this Safety Evaluation Report (SER) supplement is to demonstrate that the acquisition of core debris samples can be accomplished without presenting undue risk to the health and safety of the public.

The program will obtain up to six specimens of the TMI-2 core debris by lowering specially designed tools into the reactor which will extract samples of the loose debris and retrieve them into small shielded casks for off-site shipment and analysis. The analysis of the samples will identify the composition of the particulate core debris, determine its particle size, determine its fission product content, determine fission product leachability from the debris, analyze the drying properties of the debris, and further, determine whether or not pyrophoric materials exist in the core debris.

2.0 Description of Core Debris Sampling

The tasks associated with obtaining samples of the TMI-2 core debris will be performed in accordance with detailed, approved procedures. The synopsis provided below is intended to give a general overview of these activities.

With the reactor coolant system depressurized and the water level lowered below the CRDM closures, access to the core cavity and the debris bed is available. It is intended to obtain up to six samples of the core debris by lowering tools into the reactor that will grab samples of loose debris and retrieve them into a small shielded cask for off-site shipment.

Six debris sample tools and their necessary support equipment will be staged on the reactor service structure. A small core debris sample cask will be attached to the top of the H-8 manipulator tube. A core debris sampling tool will then be lowered through the cask and down the tube to the top of the core. When the tool is just above the surface of the core debris, the sampler will be opened and the tool will be lowered into the debris to obtain a sample of approximately one(1) cubic inch as confirmed by testing. The debris sample will then be lifted into the cask and the deployment boom will be uncoupled and the cask sealed. The debris sample will be double contained at this point (within both the sampler and the transfer cask). During the sample lifting operation, the radiological control technician will monitor and record the gamma dose rates in the vicinity.

The above process will be repeated using two other transfer casks and debris sampling tools which will obtain samples from approximately three to six (3-6) and ten to fourteen (10-14) inches below the top of the rubble bed, rather than from the surface of the bed. Three additional samples (one surface and two sub-surface) will be obtained at core location E9.

The six samples, inside their respective casks, will be transferred from the CRDM service structure to the personnel airlock. They will then be taken through the airlock and placed in their shipping containers mounted on a vehicle outside the airlock. At present, three types of shipping containers are being considered (a) N-55 overpak with the sample contained in a 17-H drum (b) a 20 WC overpak with the sample in a 2R container or (c) a CNS-113-C cask with the sample in a 2R container. Final decisions will be made at a later date. Shipping plans will be handled separately.

3.0 Justification for In-Containment Task Completion Milestone

The analysis of particulate core debris from TMI-2 will benefit both the reactor recovery and data acquisition tasks as follows:

Reactor Recovery

o Particulate debris removal system

As part of the defueling process, the fuel debris will be packaged (e.g., into canisters) and immobilized for off-site shipment to a DOE facility. Several important properties of the particulate debris must be known for the design of a system to process the debris. Design of debris removal and separation equipment requires a knowledge of the core debris particle size distribution. Particle size data are also needed so that the proper particle trapping equipment (settling tanks, cyclones, centrifuges, etc.) can be selected. Knowledge of the size fraction $< 100\mu$ will aid in estimates of vessel water turbidity changes during particulate removal. Data on the radionuclide curie content as a function of particle size will permit better estimates of the radiation source term. The chemical composition of the particulate debris must also be analyzed. Identification of the relative amounts of fuel, cladding, structural and control materials and their particular densities are needed. Also, it is known that some metal hydroxides are gelatinous and will readily plug filters and reduce their capacity. Since such compounds could have formed from reaction of the stainless steel in the core, chemical analysis of the debris to be dried at moderate temperatures will be investigated. If the particulate debris has chemically bound water (water of hydration) then radiolytic gas generation may severely complicate off-site shipping of the material. This condition was encountered in preparing zeolite SDS liners for shipment.

o Canal Water Cleanup System

The effluent water from the particulate debris removal system may be directed to one or more canal water cleanup systems to remove soluble fission products. The selection of demineralizers and the amount of demineralizers needed will depend upon the tendancy of the particulate debris to release fission products (particularly cesium) as it is agitated during pumping or vacuuming. Specimens of debris obtained during grab sampling can be subjected to standard leachability tests to determine the range of the water source term that can be expected.

- o Pyrophoric Materials Analysis

  Analysis of grab samples will determine whether or not potentially pyrophoric materials (e.g., zirconium hydride, partially unoxidized zircaloy fines) exist in particulate debris.

Data Acquisition

Analysis of grab samples may answer present technical speculation on certain aspects of severe fuel damage accidents. These include:

- The extent of UO_2 oxidation
- Residual hydrogen in the zircaloy cladding
- The chemical and physical form of the Ag-In-Cd
- The extent of stainless steel melting
- The chemical form of certain fission products

4.0 Radiological Considerations

4.1 External Exposures

All individuals entering the reactor building will be monitored for external exposures in accordance with radiological control procedures to ensure personnel exposures are maintained within 10 CFR 20 dose equivalent guidelines. Administrative control points in accordance with the procedures will be used in order to assure specified dose limits are not exceeded. Extremity monitoring will be performed in accordance with existing procedures. Monitoring of personnel radiation exposure will be conducted throughout the execution of this task. The task will be performed with the airlock doors secured to maintain reactor building isolation.

The samples will be obtained at any time after access to the reactor vessel is available, but not simultaneous with other reactor related operations. With the RCS at Elevation 321'-6", the greatest potential for increased radiation levels at the working area of the service structure exists. As described in the SER for Underhead Characterization, the increase in the dose rate above normal background at the top of the service structure is expected to be approximately 20 mRem/hr.

Preliminary calculations were performed to determine estimated exposure rates during the grab sample task performance. Exposure rate calculations were performed assuming the fuel in the sample was exposed during reactor operation to the core average neutron flux. Two cases were analysed a) assuming no cesium leaching b) assuming 40% cesium leaching. Calculations further assumed the sampler was full of debris (39.3 cc) with a volume void fraction of 0.3. The sample, therefore, was assumed to consist of the following:

Material	Volume Fraction	%
Fuel (UO_2)	0.303	70.3
Zircaloy 4	0.102	23.7
Stainless Steel	0.003	0.7
Control Rod Material	0.023	5.3

The steel thickness for shielding consisted of 2.25 inches (5.715 cm) of steel cask wall and 0.065 inches (0.165 cm) of sampler wall. The I SØSHLD-II computer code was used to determine exposure rates at various distances from the cask. The following is the results of these calculations:

	1 Foot	2 Feet	3 Feet
Case (a) 100% Cs present	395 mR/hr	120 mR/hr	57 mR/hr
Case (b) 60% Cs present	252 mR/hr	77 mR/hr	36 mR/hr

The contribution to the above radiation levels from activation of materials within the core was found to be negligable. Therefore, assuming four-six man teams to stage and unstage the equipment and three-four man teams to retrieve the samples, it is estimated that the total exposure will be 8.4 person rem. Due to the uncertainties associated with this estimate, 10 ± 5 man-rem has been selected as the estimate for the performance of the debris sample program.

In the course of moving the sample from the debris bed to the cask at the top of H-8, it was further assumed that the sample became stuck in the manipulator tube of the H-8 location. The cask is provided with a steel skirt which fits over the manipulator tube and extends down from the top of the tube to the CRDM seimsic plate. Therefore, it was postulated that the specimen stuck at a location two feet below the CRDM support structure grating to eliminate any shielding contribution from the cask and skirt. The calculated radiation field two feet from an unshielded debris sample was 1.15 R/hr.

Therefore, it has been concluded that should the sample become stuck, protection to the operators is provided by distance and procedures will instruct them to step away from the sample position should this occur. Procedures will also specify putting the sample back into the core cavity should the radiation field exceed 75 R/hr as measured at the sampler cask support with the sample at the approximate elevation of the detector. The radiation from the specimen will be closely monitored as it is raised into the cask.

4.2 Handling Concerns

After the specimens have been retrieved from the debris bed, they will be left in their respective casks atop of the CRDM service structure or individually moved to the 347' elevation utilizing the 5-ton hoist. After all specimens have been obtained, the six casks will be moved to the 305' level at the entrance to the airlock. The casks and other hardware will be transported utilizing the RB crane (if missile shields are removed) or a combination of the five ton RB crane attachment, the missile shield trolley and hoist, and two man carrying teams.

If the missile shields have not been moved, the debris sample casks will have to be transferred from the CRDM service structure manually by a two man team or by rigging from the 5-ton hoist. The most difficult traverse in moving the cask to the 347' floor is to and from the catwalk above the fuel pool if the 5-ton crane cannot be used. During this traverse, the cask could fall from the catwalk to either the service structure or the 347' floor. In either case, the cask might drop approximately ten(10) feet. A cask drop test was performed with the cask loaded with simulated particulate core debris from a height of ten(10) feet onto a concrete surface. The cask was in a poly bag. After dropping the cask, no debris was observable in the poly bag. Consequently, it is assumed that the cask will not release any significant radioactive material, should it be dropped in the reactor building. For movement over the shallow end of the refueling canal, the floor slab can withstand a 250 lb. load drop provided the dropped object has a contact area with equivalent diameter greater than 1" and the lift height is restricted to 110 ft. For movement over the 347'-6" elevation, the concrete floor slab can withstand a 250 lb. load drop provided the dropped object has an equivalent diameter greater than 1" and the lift height is restricted to 85 ft. For movement over the 305' elevation, the 3' thick concrete slab west of the hatch, can withstand a 250 lb. load drop provided the dropped object has a contact area with an equivalent diameter greater than 1" and the lift height is restricted to 128 ft. The postulated drop of a cask will lie within these boundaries due to local deformation of both the dropped object and the concrete slabs. Further, a qualitative evaluation of the relative energies of the calculated enveloping cases stated above with respect to the energy available from a drop of the cask onto the concrete floor slabs leads to the conclusion that the integrated damage potential is well within the calculated bounds for structural failure and, hence does not pose an unacceptable risk.

4.3 Criticality

4.3.1 Recriticality of the Core

The core debris sample program involves inserting a sampling tool through the manipulator support tube and CRDM housing, through the plenum and into the rubble bed. This raises the potential for disturbing the core. This has been evaluated and the consequences are considered no more severe than the probe of the rubble bed performed as part of "Quick Look".

The probing of the core rubble bed during "Quick Look" was within the bounds of the criticality analysis described in Babcock & Wilcox Report BAW-1738, which was submitted to the NRC as part of the safety evaluation for the "Quick Look" program. Since the potential core disturbances associated with debris sampling are considered no more severe than the "Quick Look" probe which penetrated the debris bed approximately 14 inches, it is concluded that the consequences of potential core disturbances are bounded by BAW-1738 and the SER for the APSR Insertion Test.

During the core sample debris program, the RCS boron concentration will be maintained greater than or equal to 3500 ppm, which based on BAW-1738, will ensure subcriticality.

4.3.2 Criticality of Sample

The sample to be obtained from the debris bed will be retrieved in a container whose volume is approximately 2.4 cubic inches (39.3 cc), but will remove only about one cubic inch as demonstrated by test. If the container were filled with fuel and unborated water with the optimum fuel to water ratio, it is too small to constitute a critical mass. This evaluation is supported by information contained in both Savannah River Publication DP-1014 and Lynchburg Commercial Fuel Plant SNM-1168. The former indicates that 28000 cc of 3% enriched UO_2 in the optimum configuration in H_2O is required to achieve criticality. The latter indicates that 14000 cc of 4% enriched UO_2 in the optimum configuration in H_2O is required to achieve criticality. Consequently, one specimen could not represent a critical mass, nor could all six specimens if combined together.

4.3.3 Measures Taken to Reduce Operational Exposure to As Low As Reasonably Achievable (ALARA) Levels

The measures taken to minimize occupational exposure will be the same as those used in the Underhead Characterization Program.

4.4 Programmatic ALARA

Programmatic ALARA has been considered for the accomplishment of the task. The core debris samples are considered necessary at this time to provide data which will influence defueling tooling and water cleanup systems. This influence will significantly reduce exposure to workers during the plenum/fuel removal phases of the overall cleanup program, as demonstrated by the modifications necessitated to the reactor vessel head removal task due to the data obtained from the first "Quick Scan" program.

5.0 Pyrophoric Concerns

The subject of a possible pyrophoric reaction is addressed in the "Safety Evaluation Report for Radiation Characterization Under the Reactor Vessel Head." The discussion of this concern in that SER was primarily devoted to the possibility of a pyrophoric reaction of debris on the plenum cover. For the core debris sampling program, the debris in question will be that of the actual core; however, the potential for a pyrophoric reaction of the sample is still considered highly unlikely for reasons outlined below.

- As evidenced by various burning of zirconium fires, the rapid burning of zirconium metal is usually restricted to fines 100 μm or less. The formation of zirconium powder during the accident is extremely unlikely due to the dynamics of the accident.

- Fines that might be in the sample would be diluted with other fully oxidized and non-pyrophoric materials which would tend to inhibit sustaining a pyrophoric reaction, as evidenced by examination of the material from the surface of the leadscrew.

- Debris from the CRDM leadscrew which was removed during the "Quick Look" did not ignite during various tests which attempted to burn it. Other samples of residue material from the reactor coolant system which have been extensively handled and examined in air have failed to exhibit any observable pyrophoric properties.

Although the possibility of withdrawing pyrophoric material from the debris bed is small, the possible consequences of having such material in the debris sample were considered. The nominal size sample being withdrawn is approximately one(1) cubic inch, as confirmed by testing. Any debris on the external parts of the sampler is expected to be washed off during withdrawal and need not be considered. If this sample contained 10% zirconium (a fuel assembly is approximately 10% zirconium by volume) and if it were to totally burn when exposed to the air, 123 BTU of energy would be released. (An explosion of this material is not considered possible.) This heat would, assuming perfect heat transfer, raise the specimen holder approximately 200°F. This is insufficient to cause damage to either the holder, the manipulator tube, or the sample cask. Therefore, it is concluded that a possible pyrophoric occurrence will not increase the radiological and safety evaluations considered for the debris sample program.

6.0 Additional Safety Concerns

A review of the activities associated with the debris sample program has not revealed any additional safety considerations not previously considered in the Underhead Radiation Characterization Program. The following potential concerns were also considered:

* decay heat removal
* hydrogen evolution
* boron dilution

The debris sample program consists of activities similar to those performed during the core probe into the debris bed which was safely performed previously. It is, therefore, concluded that the debris sample program will not present any undue risk to the health and safety of the public.

10 CFR 50, Paragraph 50.59, permits the holder of an operating license to make changes to the facility or perform a test or experiment, provided the change, test, or experiment is determined not to be an unreviewed safety question and does not involve a modification of the plant technical specifications.

A proposed change involves an unreviewed safety question if:

a) The possibility 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

b) The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or

c) The margin of safety, as defined in the basis for any technical specification, is reduced.

The planned activities will not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated. This is based on the work being performed in accordance with approved procedures, measures to be taken for the prevention of an RCS boron dilution event, and that potential disturbances of the core are within previously evaluated bounds.

The debris sample program will not create the possibility of an accident or malfunction of a different type than any evaluated previously. This is based on the review of potential accidents identified in this and previously submitted SER's.

The tasks included in this SER will not reduce the margin of safety as defined in the basis for any technical specification. This is based on operating systems and equipment covered by the technical specification in accordance with approved procedures. Also, the releases of radioactivity to the environment have been shown to be within techincal specification limits.

Therefore, it is concluded that the acquisition of core debris samples does not involve any unreviewed safety question as defined in 10 CFR Part 50, Paragraph 50.59.

7.0 Conclusion

Based on the radiological and safety evaluations considered for the core debris sample program, the activities may be accomplished without presenting undue risk to the health and safety of the public and employees.