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May 25, 1983
4410-83-L-0098

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

Dear Sir:

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

Attached for your review and approval is Revision 0 to the Safety Evaluation Report for the Underhead Characterization Study to be performed. This Study will include the following activities: (Polar Crane not available)

- Raising and lowering the RCS level
- Obtaining dose rates around the head and service structure
- Inserting an ionization chamber into CRDM H-8
- Performing a visual inspection under the reactor head with remote equipment
- Inserting TLD's
- Obtaining debris samples from the top of the plenum

This Safety Evaluation also covers activities performed in support of the activities identified. The Safety Evaluation differs from the Safety Evaluation Report previously submitted via 4410-83-L-0040 in that no credit is taken for use of the Polar Crane as the Polar Crane may not be available in time to support this evolution. Instead, all activities will be performed with the missile shields in their current position.

This activity is currently scheduled to start in mid June, 1983.

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

Sincerely,

B. K. Kanga

B. K. Kanga
Director, TMI-2

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BKK/RBS/jep
Attachment
CC: Dr. B. J. Snyder, Program Director - TMI Program Office

Safety Evaluation Report

for

Radiation Characterization Under the Reactor Vessel Head
(Polar Crane Not Available)

May 1983

Three Mile Island
Unit 2

Revision 0
May 17, 1983

SAFETY EVALUATION REPORT
FOR
RADIATION CHARACTERIZATION
UNDER THE REACTOR VESSEL HEAD

SAFETY EVALUATION REPORT
FOR
RADIATION CHARACTERIZATION
UNDER THE REACTOR VESSEL HEAD

1.0 PURPOSE, SCOPE, AND ORGANIZATION

1.1 PURPOSE

The purpose of this Safety Evaluation Report (SER) is to demonstrate that the evolutions comprising radiation characterization under the reactor vessel head can be accomplished without presenting undue risk to the health and safety of the public.

The objective of this characterization program is to ensure that adequate radiological protection measures will be taken to keep radiation exposure as low as reasonably achievable (ALARA) during head removal. To achieve that objective, it is necessary to make measurements with the reactor vessel head still in place. These measurements need to be made in a manner that will allow the following:

- A determination of the radiation source contributors under the head and the approximate dose rate contribution from each

- A conservative estimate of radiological conditions that will accompany head lift

- Identification of engineered features needed (if any) to render head lift radiologically acceptable

A multi-phase approach has been selected. The first phase consisted of a simple vertical scan of the underhead region with an ionization chamber. This "Quick Scan" resulted in measured radiation levels that were higher than previously assumed. Consequently, a more detailed measurement program is to be implemented.

1.2 SCOPE

This safety evaluation is applicable to the next phase of the characterization program. The major activities involved in the underhead radiation characterization are:

- raising and lowering the reactor coolant system (RCS) water level,

- obtaining dose rates at various points around the head and service structure,

- pulling one or more control rod drive mechanisms (CRDMs) and venting all others

- inserting an ionization chamber into the opening created by the removal of a CRDM

- performing visual inspection under the reactor pressure vessel head with a TV camera,
- inserting a string of thermoluminescent dosimeters (TLD), and
- obtaining a debris sample from the top of the plenum.

Also covered by this safety evaluation are those activities performed in direct support of the major activities identified.

1.3 ORGANIZATION

Section 2 of this SER briefly describes the major activities associated with the underhead characterization.

Section 3 discusses the radiological aspects of the activities involved in the characterization program.

Section 4 of this SER identifies the safety concerns associated with the performance of the characterization program and addresses each of the concerns.

Section 5 provides a 10 CFR 50.59 evaluation for underhead characterization and Section 6 summarizes the conclusions of the safety evaluation.

2.0 DESCRIPTION OF THE UNDERHEAD CHARACTERIZATION PROGRAM

2.1 SYNOPSIS

The tasks associated with the underhead characterization program will be performed in accordance with detailed, approved procedures. The synopsis provided below is intended to give a general overview of the activities.

The reactor coolant level in the reactor vessel will be lowered to below El. 345' and all of the CRDM's will be vented. The coolant level will then be lowered to approximately El. 324'6". One or more CRDM's will be removed and dummy motor tubes (also called the manipulator support tube) will be installed on the CRDM nozzle flanges. It is planned to use a hoist/trolley system located on top of the missile shields. This will be used to lift the CRDM stator/position indicator assembly and the CRDM with the attached leadscrew support tube (LSST). Sufficient clearance exists between the service structure platform and the bottom of the missile shields to allow cutting of the LSST at the CRDM flange. Once the LSST is detached and secured the CRDM is either lowered through the service structure pendant hatch and secured, or secured on the service structure platform. The hoist will then raise the LSST to allow a bottom segment of the LSST to be cut for future examination. This specimen will be used to characterize the stainless steel in the underhead environment. Shielding will be provided, if necessary, to keep radiation levels at the 347' elevation as low as reasonably achievable.

However, if at any time during the lifting of the CRDM, should the area radiation monitor at the perimeter of the service structure platform register 2 rem/hr or greater, or should the radiation monitor at the CRDM flange elevation register 100 rem/hr, the CRDM will be lowered back to its original position. The stator/position indicator assembly of two adjacent CRDM's would then be removed. The CRDM would again be raised until the bottom of the LSST clears the top of CRDM flange on the reactor vessel head and then translated to the areas vacated by the two adjacent stators. The CRDM would be secured at the service structure platform. This procedure, if required, would allow the installation of the manipulator support tube, but would not allow a sample of the LSST to be taken.

A camera can then be used to obtain CCTV pictures of the top of the plenum and the underside of the reactor vessel head. Depending on the results of the visual inspection, an attempt will be made to remove a sample of debris from the top surface of the plenum.

With the RCS water level still at approximately El. 324'6", the Quick Scan radiation survey will be repeated. This involves taking radiation readings at various elevations using an instrument lowered through the manipulator support tube. These surveys will provide verification of the first Quick Scan and additional information regarding the plenum and core environment. Radiation monitoring instruments will be installed or surveys taken:

- o on the reactor head above the studs,
- o inside the service structure, between selected CRDM housings, as close as possible to the CRDM flanges,
- o at the work platform on the service structure, and
- o above an open CRDM.

This will provide information on the increase in radiation levels around the head and service structure when the RCS water level is lowered.

A string of TLD's will also be lowered into the reactor vessel to obtain data on radiation levels.

Next, the RCS water level will be lowered to El. 321'-6". Radiation readings will be taken both inside the reactor vessel head and outside the head service structure as the water level is reduced. With the RCS water level at El. 321'-6", the video inspection will be repeated. If it is deemed necessary, another debris sample will be obtained. Radiation data will again be obtained, as described above for the water level at El. 324'6".

2.2 REACTOR COOLANT PRESSURE BOUNDARY (RCPB)

Removal of one CRDM and venting of the remainder of the CRDM's (total of 69) will result in the permanent cessation of the Reactor Coolant Pressure Boundary. For reasons provided below, a pressure retaining boundary is no longer required for the Reactor Coolant System.

The design purpose of the RCPB is to protect the health and safety of the public from design basis accidents with the plant in operational modes 1 through 5. During underhead characterization activities, the reactor will be in its current shutdown condition. In this condition, the functions of the reactor coolant system are to remove decay heat and to retain boron concentration for reactivity control. Maintenance of the RCPB is not necessary for the reactor coolant system to perform its shutdown mode functions as discussed in Sections 4.2 and 4.3.

For underhead characterization activities, no events have been postulated to occur which could cause or require the RCS to become pressurized; therefore, the RCPB is not required during underhead characterization activities.

Nevertheless, certain additional safety measures will be taken to mitigate the consequences of potential unanticipated occurrences. In this regard, the term "core alteration" shall be defined as the movement or manipulation of fuel within the reactor pressure vessel with the reactor coolant pressure boundary open to the containment atmosphere, or the movement or manipulation of other components within the reactor pressure vessel which could reasonably result in a disturbance of the fuel configuration with the reactor coolant pressure boundary open to the containment atmosphere.

During the evolutions of underhead characterization which may result in core alteration or movement of irradiated fuel within the containment, the following precautions shall be observed:

(1) The equipment hatch closed.
(2) A minimum of one door in each airlock closed.
(3) Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:
    (a) Closed by an isolation valve, blind flange, or manual valve, or
    (b) Be capable of being closed by an operable automatic containment purge and exhaust isolation valve.

If any of the above conditions cannot be met, all operations involving core alteration or movement of irradiated fuel in the containment building shall be immediately suspended. Suspension of core alteration shall not preclude completion of movement of fuel or other components to a safe, conservative position.

Following underhead characterization activities, the reactor coolant system will remain open to the containment atmosphere in anticipation of head removal. This condition precludes the ability to repressurize the RCS. However, for the reasons cited above, pressurizaton of the RCS is no longer required.

2.3 LIFTING DEVICES

As stated in Section 2.1, a hoist located on the missile shields will be used for all lifts associated with the underhead characterization. The hoist/trolley assembly is rated at 1200 pounds with the hoist rated at 1.5 tons. The assembly will be load tested at 1800 pounds prior to use inside containment. The items to be lifted, their maximum weights, and their respective load paths are listed in the following table:

Item	Maximum Weight (lbs.)	Load Path
Stator	500	Vertical lift to secured position on service structure platform
CRDM	800	Vertical lift to secured position on service structure platform
LSST	30	Vertical lift to service structure platform
Manipulator Support Tube	200	From floor of el. 305' to service platform
Tools	500	From floor of el. 305' to service platform

3.0 RADIOLOGICAL CONSIDERATIONS

3.1 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.

The underhead characterization requires that the RCS water level be lowered to El. 321'-6", which is approximately 1' below the top of the plenum. This lowering of the water in the reactor vessel removes the shielding provided by the water. This raises the potential for an increase in radiation levels in the area of the head and service structure. An evaluation of the potential increase in radiation levels resulting from lowering the RCS water level is presented below.

The increase in dose rates around the reactor vessel head are expected to be relatively small. This is based on the reactor vessel head, which has a minimum thickness of approximately 6.75", providing an attenuation factor of about 7×10^{-4} for radiation sources from within the reactor vessel. Currently, dose rates around the outside of the reactor vessel head are approximately 150 to 300 mrem/hr due largely to contamination of surfaces external to the reactor vessel. The contributions from the deposited material and liquid inside the vessel are neglible. Estimates of dose rates on the underside surface of the reactor vessel head, with the RCS water level lowered, range from 200 to 600 Rem/hr. This would result in an increase of approximately 140 to 420 mrem/hr on the outside of the reactor vessel head.

The increase in the dose rate on the service structure platform resulting from lowering the RCS level will be much less than the increase adjacent to the vessel head. This is due to the combined effects of extra distance and shielding. The top of the service structure is more than 20 feet above the vessel head. This will account for a substantial reduction in the degree of dose rate increase. Also, the service structure houses 69 CRDM's in a tightly spaced configuration. The CRDM's constrict the pathway of radiation emanating from the vessel head so that only a fraction of the space above the head is available for streaming. The result is that the dose rate increase on the service structure platform resulting from a lower RCS level is considered to be acceptably low.

The increase in radiation levels over a removed CRDM has been estimated using the results from the Quick Scan performed in December 1982. Since the CRDM penetration through the reactor vessel head and service structure will act as a collimator, only a portion of the surface below the reactor vessel head will contribute to the dose rate over the removed CRDM. The analysis was performed assuming a conservative plateout source of 25,000 $\mu Ci/cm^2$ of Cs-137 on the top of the plenum. The result of the analysis was an increase in the dose rate above background levels over the removed CRDM of less than 20 mrem/hr. Contributions from vented CRDM's would be neglible compared to this. The present general area dose rate on top of the service structure is 50 to 150 mrem/hr. The increase in the general area dose rate on the top of the service structure of less than 20 mrem/hr is relatively small.

Based on the above, the potential increases in general area radiation levels resulting from lowering the RCS water level and removing a CRDM are considered acceptable. Measurements will be taken for varying water level conditions to assess the actual conditions with respect to the removed CRDM.

The total exposure for the underhead radiation characterization is estimated to be 16 person-rem. This is based upon the scope defined in Section 1.2 of this safety evaluation. This estimate is based on the majority of the person-hour estimate being in a dose fielo of approximatley 100 mrem/hr and approximately 7.4 person-hours in higher dose fields during the actual CRDM lift, the cutting of the LSST, and the retrieval of the camera. This estimate would be reduced by approximately 1 person-rem should the "fall-back" procedure, detailed in Section 2.1, be used. Due to the uncertainty in both the person-hour estimate and the dose, it is estimated that the total exposure could vary by up to $\pm$ 50 percent. Considering the uncertainties associated with the person-rem estimate, 8 to 24 person-rem has been selected to be used as the estimate for the performance of the underhead radiation characterization.

3.2 INTERNAL EXPOSURES

The ongoing decontamination of the Reactor Building has been effective in reducing the airborne activity. Since the underhead characterization program will not increase the airborne activity level in the Reactor Building as shown in Section 4.4, the safe and manageable condition of airborne activity will persist.

As specified by radiological control procedures, analyses of expected airborne activity levels will be performed in order to select appropriate respiratory protective devices for personnel entering the Reactor Building. These devices are used to protect against particulate radioactivity. Other forms of radioactivity, such as noble gases and tritium, are not expected to pose difficulties. Current data from air samples of the Reactor Building indicate a mean tritium airborne concentration of 1.0 E-6 $\mu Ci/cc$ which is equivalent to 0.20 mpc-h per hour.

3.3 MEASURES TAKEN TO REDUCE OCCUPATIONAL EXPOSURE TO AS LOW AS REASONABLY ACHIEVABLE (ALARA) LEVELS

The objective of minimizing occupational exposure has been a major goal in the planning and preparation for all activities in the containment. The actions that have been taken or are being planned toward meeting this objective are summarized in this section. Protective clothing and respirators will be used as necessary to reduce the potential for external contamination and internal exposure of personnel.

Execution of individual tasks are maintained ALARA by a detailed radiological review by Radiological Engineering and mockup training of work crews. This training will approximate the actual work situation as closely as can be achieved for each task utilizing appropriate equipment, protective clothing, and respiratory protection.

Extensive planning of tasks to be conducted in a radiation field, and training of personnel will be used to reduce the time needed to complete a task. Extensive use of photographs and the in-containment closed circuit television system will be used to familiarize personnel with the work area. The higher radiation areas are identified to personnel and shielded where practicable. The work is structured to avoid these areas to the extent possible. All personnel assigned to the various tasks are trained to respond to unforeseen radiation levels. As stated in Section 2.1, should a radiation level exceed a prescribed limit, personnel will have been trained to cease present activities and proceed to the "fall-back" procedure. Should a radiation level limit still be exceeded, while performing the "fall-back" procedure, the CRDM will be lowered back to its original position and all subsequent tasks will be halted. Practice sessions will be utilized as necessary to ensure that personnel understand their assignments prior to entering the containment. Planning and training are proven methods of ensuring that personnel are properly prepared to conduct the assigned task expeditiously.

4.0 SAFETY CONCERNS

4.1 GENERAL

A review of the activities associated with the underhead radiation characterization identified the following issues:

- decay heat removal,
- criticality,
- release of radioactivity,
- hydrogen evolution, and
- pyrophoricity

Each of these issues is discussed below.

4.2 DECAY HEAT REMOVAL

As noted in Section 2.1, the performance of the underhead radiation characterization requires that the RCS water level be lowered to approximately El. 321'-6", which is about 1' below the top of the plenum. At El. 321'-6", there is significantly less water volume in the RCS than has been maintained in the past. As a result of having less water volume in the RCS, the ability to continue to adequately remove decay heat and maintain the bulk RCS temperature within acceptable limits for the losses to ambient cooling mode ($\leq$198°F) was investigated.

An analysis of decay heat removal ability with the RCS water level at El. 323'-6" (approximately 1' above the top of the plenum) was submitted to the NRC as an appendix to the Quick Look safety evaluation (Reference 1). An additional analysis (Reference 2) has been performed with the RCS water level at El. 321'6" and at the bottom of the reactor vessel nozzles (El. 314'). The results of the new analysis show that the expected rise in RCS temperature will be acceptable.

Reference 2 presents both a conservative analysis and a best estimate analysis. The conservative calculations were made with the models originally developed for the appendix to the Quick Look safety evaluation. This conservative analysis resulted in RCS bulk temperatures of 165°F and 198°F for an RCS water level at El. 321'-6" and 314', respectively. These temperatures were based on the decay heat rate for December 1, 1982. The analysis shows that the RCS bulk temperature decreases with time for the same water level as the decay heat rate decreases.

The best estimate models, benchmarked to temperatures measured following the partial draindown for the Quick Look inspection, were developed and used to predict the expected RCS bulk temperatures. These models resulted in RCS temperatures of between 112°F and 120°F with the RCS water level at El. 321'-6", and 149°F to 152°F with the RCS water level at El. 314'. These temperatures are based on the decay heat rates for December 1, 1982 and July 1, 1983, respectively.

Subsequent to the Quick Look, a test instrument was inserted into the reactor vessel to a point above the rubble bed to verify the temperatures being recorded by the incore thermocouples. The results of this effort indicate that the incore thermocouples are providing a reasonable measurement of the temperature of the water in the core cavity. These results were transmitted to the NRC by Reference 3.

Since test results indicate that the incore thermocouples are providing reasonable measurements of the temperature of the water in the core cavity, and the best estimate models are benchmarked against the incore thermocouple readings, there is reasonable assurance that the temperatures predicted by the best estimate models are accurate. Therefore, it can be concluded that the bulk RCS temperature for the draindown conditions associated with underhead characterization are well within the acceptable values for the present losses to ambient cooling mode.

4.3 CRITICALITY

4.3.1 Potential Core Disturbances

The underhead radiation characterization program involves inserting an instrument through the manipulator support tube, through the plenum, and down to the rubble bed. This raises the potential for disturbing the core. This has been evaluated and the consequences of the instrument contacting the rubble bed, either intentionally during the insertion or inadvertently for any reason, are considered to be no more severe than the probe of the rubble bed performed as part of the Quick Look.

The probing of the rubble bed during the Quick Look was within the bounds of the criticality analysis described in Babcock and Wilcox Report BAW-1738, which was submitted to the NRC as part of the safety evaluation for the Quick Look (Reference 1). Since the potential core disturbances associated with these data acquisition tasks are considered no more severe than the Quick Look core probe, it is concluded that the consequence of potential core disturbances are bounded by BAW-1738.

During the underhead characterization activities 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 Reactor Coolant System (RCS) Boron Dilution

4.3.2.1 Introduction

During the underhead characterization phase of recovery, reactor shutdown (subcriticality) will be assured by the presence of boron in the reactor coolant. The Safety Evaluation for Insertion of a Camera through a Leadscrew Opening stated that maintaining RCS boron concentrations of 3500 ppm or greater assures sub-criticality under all credible conditions. This is supported by a review of the information obtained during axial power shaping rod insertion and the Quick Look.

TMI Unit 2 operations during the past four years have demonstrated that it is possible to maintain a controlled boron concentration in the RCS. Recently, during the Quick Look testing, the RCS conditions differed from those that existed during the previous years. The primary coolant level was lowered and the primary coolant pressure reduced. In the underhead characterization phase of recovery, the reactor coolant water level will be lowered below the Quick Look level (elevation range of 331' to 335'). The level will be lowered to approximately one(1) foot below the reactor vessel flange i.e., Elevation 321'-6". In view of these differences, it is necessary to evaluate the ability to continue to reliably maintain a controlled boron concentration in the RCS. The purpose of this section is to review the precautions that will be taken to assure that the required RCS boron concentration will be maintained.

The RCS temperature and chemistry will not be significantly affected during the underhead characterization; hence, boron solubility will remain essentially unchanged. In fact, the slight increase in RCS temperature discussed in Section 4.2 on Decay Heat Removal will improve boron solubility conditions.

The only way RCS boron concentration can be changed in an uncontrolled manner during underhead characterization is by dilution of the RCS coolant with water that is either unborated or borated below 3500 ppm.

The following discussion shows that the procedures in effect while the RCS is depressurized will prevent the uncontrolled addition of coolant to the RCS, and hence, prevent the uncontrolled reduction of the boron concentration. In addition, if for some unforeseen reason boron dilution should occur, the monitoring and corrective action procedures will preclude significant reductions in boron concentration and assure the reactor remains subcritical.

4.3.2.2 Actions Taken to Prevent Boron Dilution

As stated above, boron dilution will result if water containing boron concentrations less than 3500 ppm is added to the RCS. The potential sources of this water are the various systems connected to the RCS, which includes the secondary system. Systems which potentially contain coolant with boron concentrations less than 3500 ppm have been reviewed and isolated to assure that they will not be credible sources of boron dilution. Two isolation boundaries are provided for each potential inleakage path. An isolation boundary is defined as a closed tagged-out valve, electrically locked out pump, removed spool piece, heat exchanger tube boundary, or pressure differential.

The following actions will be taken to prevent the unintentional dilution of the boron in the RCS. It is concluded that these actions will prevent the dilution of the RCS boron concentration during the time the pressure and water level are lowered.

a. <u>Steam Generator</u>

One potential source of dilution of the RCS boron is secondary coolant leakage through the steam generator tubes. The potential for this leakage has been precluded in the past by maintaining the RCS pressure higher than the secondary cooling pressure. Thus, any leakage would be from the primary system to the secondary system.

During the underhead characterization, the reactor vessel water level will be subjected to building pressure while the hot legs and pressurizer will be under a nitrogen blanket of approximately one(1) psig. To preclude RCS dilution, procedures require that water levels and cover pressures in the secondary side of the steam generators be maintained lower than those in the primary side.

The secondary side of the once through steam generator (OTSG) will be drained to below 313'-0" and the upper voided portion will be filled with nitrogen to a pressure of one ($\pm$ one) psig. The reactor vessel water level will be lowered to a minimum level of 321-3". (This accounts for a 3" tolerance in the RCS level indication.) This results in a minimum level in the hot leg or primary side of the steam generators of 318'-7". The nitrogen pressure above the primary side of the OTSG's will be maintained at one psig. Therefore, as a minimum, there will exist a small pressure difference across the OTSG tubes which would cause flow, should a leak occur, to be from primary to secondary. At all reactor vessel water levels above 321'-3", the pressure differential will increase.

To monitor possible water leakage into the OTSG's, each generator is equipped with a level measuring device. The "A" OTSG incorporates a pressure gauge at the 281' elevation in the auxiliary building and a standpipe in the service building and the "B" OTSG has a water filled tygon tube in the reactor building.

In addition, possible inleakage paths will be isolated. Surveillance of the levels and valve positions are performed on a periodic basis as required by technical specifications. Under these conditions, the secondary volume of steam generators will not be a credible source of RCS boron dilution.

b. Makeup and Purification/Standby Pressure Control (SPC) Systems

These systems are borated greater than 3500 ppm and will be operated by approved procedures to letdown, process through the Submerged Demineralizer System System, and makeup back to the RCS. The makeup pumps MU-P-1A/B/C will tagged "off" and portions or connections to these systems that are not used for makeup will be isolated. Should the RCS level decrease below the controlled range, letdown from the RCS will be secured until the level increases to the controlled range. Should the level continue to decrease, makeup will be initiated from the SPC System or a bleed tank borated to $\geq$ 3500 ppm using approved procedures.

Assurance that the makeup water is borated greater than 3500 ppm will be provided by analysis of a sample taken from the appropriate Reactor Coolant Bleed Holdup tank of each batch used for makeup.

c. Demineralized Water System

The demineralized water system has been reviewed and where possible, spool pieces in the flow path to the RCS have been removed. Where this could not be done, isolation valves in the flow paths have been tagged shut.

d. Submerged Demineralizer System (SDS)

The SDS may be operated to process the water letdown from the RCS. This will not create a dilution problem because the SDS will be isolated from the Reactor Coolant System except via the appropriate Bleed Holdup Tank which will be monitored for boron content.

e. Other Systems

Decay Heat Removal
Mini Decay Heat System
Core Flood System
Intermediate Closed Cooling Water
Decay Heat Closed Cooling Water
Chemical Addition
Steam Generator Feed Water
Spent Fuel Cooling

The following actions will be taken to prevent dilution of RCS boron by unintentional transfer from the above systems of coolant containing boron concentrations less than 3500 ppm to the RCS.

i) The above systems have been reviewed and isolation valves in the flow paths have been tagged shut. The isolation criteria provides two isolation boundaries for each potential inleakage path.

ii) A checklist has been prepared listing all valves that are to be used for isolation during the time the RCS is at reduced pressure (including those in Section 4.3.2.2 b, c, and d). The position of these valves will be confirmed every 24 hours during this period.

iii) All pumps in these systems except those required to be operable per the Recovery Operations Plan and/or technical specifications will be tagged-out to further preclude the inadvertent transfer of coolant to the RCS.

iv) The levels of all storage tanks that could be sources of water into the RCS will be monitored and logged once every 24 hours.

4.3.2.3 Actions Taken to Monitor Boron Content

a. RC Level Indication

The RCS water level indication is presently available from four different instruments connected to the decay heat line external of the reactor building. This redundancy of level indication would ordinarily be sufficient to insure sufficient information to properly ascertain the correct level. However, since RCS level indication is a prime source of information to provide assurance that a boron dilution circumstance has not occurred, another level indication system will be made available. A tygon tube connected to an RCS cold leg pipe will be added. This level indication is normally used to determine RCS level in the drained down condition when the RV head is to be removed.

b. Sampling and Boron Measurements

An RCS sample will be obtained once a week as requireo by technical specification and analyzed for its boron content in accordance with approved procedures. Considering the RCS leak rate, instrument error, and sampling losses, once per week is deemed adequate.

c. Neutron Monitoring

The source range neutron instrumentation will be monitoreo.

4.3.2.4 Actions Taken to Detect and Terminate Inadvertent Boron Dilution

The actions described in Section 4.3.2.2 will prevent the inadvertent dilution of the boron in the RCS. However, even though such dilution is unlikely, procedures have been established to assure the early detection of a dilution event. In such a case, action can be taken to find the source of the dilution and stop it or to inject additional boron.

4.3.2.5 Conclusion

The actions discussed above are considered sufficient to preclude inadvertent boron dilution. In the unlikely event such dilution were to occur, procedures provide for actions which will permit its detection and provide the information needed to terminate the coolant transfer. Based upon the use of these plant limits and procedures, reactor shutdown is assured and criticality is not considereo credible.

4.4 RELEASE OF RADIOACTIVITY

The activities associated with the underhead radiation characterization were reviewed with respect to radioactive releases to the environment. It was concluded that these activities would not result in releases of radioactivity to the environment in excess of the releases described in the safety evaluation for decontamination of the reactor building (Reference 4). The basis for this conclusion is described below.

The releases to the environment presented in the safety evaluation for ongoing decontamination are based on the containment purge being in continuous operation at a purge rate of 25,000 cfm for 365 days. A source term was identified based on measurements of airborne activity in the containment atmosphere during the Decontamination Experiment and this source term was assumed to remain constant. The assumption of no reduction in the source term throughout the decontamination effort is a conservative assumption in that experience gained during the Decontamination Experiment shows a reduction in airborne activity as decontamination activities proceed. Since decontamination activities have been proceeding since the source term used for the release analysis was developed, it is reasonable to assume that it has been reduced.

The activities associated with the underhead characterization will not result in an overall increase in the airborne activity in the containment atmosphere. This is because the activities occur in areas which have had much of the loose contamination, that which is most likely to become airborne, removed by the decontamination effort. The activities do involve the opening of the reactor vessel to the containment atmosphere, which presents a potential source for additional airborne activity. However, this is not expected to increase the airborne activity in the containment atmosphere. The temperature of the water in the reactor vessel is expected to increase as noted in Section 4.2. However, since the reactor vessel head will be colder than the water, the reactor vessel head should remain wetted by condensation rather than drying out since it is located above the hot water. This will tend to suppress airborne contamination. There is no planned forced mixing of the air in the reactor vessel, although a contingency system is available to draw gas out of the vessel to prevent hydrogen buildup, if needed. Since the workers will be appropriately dressed for protection against airborne activity, worker protection is assureo.

In summary, since

- o the source term used in the calculation of releases to the environment during continued decontamination activities is conservatively high,
- o the activities are not expected to increase the airborne activity in the containment atmosphere beyond that assumed in the calculation of releases to the environment during decontamination activities, and
- o the purge rate will be the same as for ongoing decontamination,

it is concluded that the releases to the environment during the performance of the data acquisition tasks will be within the results presented in the safety evaluation for ongoing decontamination activities.

The NRC in Reference 5 concluded that the rate of release of radioactivity to the environment predicted in the safety evaluation for ongoing containment decontamination activities were within the technical specification limits of TMI-2. Since the releases to the environment will be within the results presented in the safety evaluation for ongoing decontamination activities, it is concluded that the releases, and rate of release, of radioactivity to the environment will be within the technical specification limits of TMI-2.

4.5 PYROPHORICITY

4.5.1 Introduction

In NUREG-0683 "Final Programmatic Environmental Impact Statement Related to Decontamination and Disposal of the Radioactive Wastes Resulting from March 28, 1979, Accident Three Mile Island Nuclear Station, Unit 2," Section 13.10.1.3, the NRC addressed the potential for zirconium hydride fires. The discussion assumed that operations relating to defueling would be conducted with water coverage. It was stated that zirconium hydride would not ignite under water. Consequently, there has been no potential for a pyrophoric zirconium reaction in the reactor vessel environment to date, as the reactor vessel has been completely filled with reactor coolant since the accident.

However, the underhead characterization program calls for lowering the reactor coolant level in the reactor vessel. Because some of the vessel internals (e.g., leadscrew support tubes, control rod guide assembly tubes, and upper plenum cover plate) will be uncovered and exposed to air, the issue of pyrophoricity must be addressed.

The purpose of this section is to show the preponderance of information available which leads to the conclusion that a pyrophoric event is highly unlikely.

4.5.2 Summary

For the reasons outlined below, a pyrophoric reaction is considered to be an unlikely event while the reactor coolant level is lowered.

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

- Even if such fines were formed, they would have been partially or completely transformed to zirconium oxide. This chemical form is a very stable material with no pyrophoric properties. The fines would be more oxidized than larger metal pieces due to the high surface area to volume ratio. At the least, the outer surface of any such particle would at present exhibit an oxide layer due to the oxidation which has occurred as the particles have been exposed to the reactor coolant for the past four years.

- Flow velocities during and following the accident were such that significant quantities of material of any nature would not be expected to have been transferred to the upper plenum. The results of the "Quick Look" and "Quick Scan" tests support this prediction.

- Fines that might be on the plenum 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. The leadscrew sample was obtained from a section of leadscrew which was in close proximity to the plenum cover and in the normal flow path to the cover. 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.

- During and after the "Quick Look" program, the upper tube sheets of both OTSG's were exposed to an air environment. The material on these tube sheets is expected to be similar to that which might be found on the plenum cover. No indications of a pyrophoric occurrence were evident during the four months that the tube sheets were exposed to air.

4.5.3 Evidence for Non-Pyrophoricity in TMI-2

4.5.3.1 Minimum Core Debris on the Plenum Cover

The "Quick Look" video tapes show that the top surface of the control rod guide assembly first and second support plates have only the light deposits typical of that found in normal plants. This indicates that the plenum cover should also be free of debris. The "Quick Look" tapes also show that the bottom surface of each support plate is free of deposits which indicates that there is no reason to believe there is any debris on the inside surface of the reactor vessel head.

Visual examination of the removed leadscrews indicates only a thin layer of material on their surfaces. No substantial buildup of material was observed on or between the horizontal surfaces of the threads. This observation further substantiates the premise that little core debris was carried to the plenum cover during or after the accident.

This conclusion appears reasonable based on the flow conditions estimated to exist at the time of the accident. In particular, the principal means by which debris could reach the plenum top cover and inside surface of the vessel head is by entrainment in fluid flowing upward inside of the control rod guide assemblies. It should be noted that this bypass flow was a small fraction of the total flow. With one reactor coolant pump running such as occurred after the March 28, 1979 accident, the vertical velocity within a guide assembly is estimated to be in the order of 0.3 feet per second in the region between support plates. This velocity is low enough to permit most of the entrained fuel debris to settle out before it could reach the top end of a guide assembly. Only small particles, on the order of 10's of microns in size or less, could reach the upper end of the guide assembly. Because of their small size and the core conditions that resulted in their formation, any particles that did reach the upper plenum surface are likely to be completely oxidized.

Quick Scan experiment results concluded that the activity levels measured in the upper plenum were most likely a result of cesium deposition on all vertical and horizontal surfaces. Calculations show that it is unlikely that the activity levels are a result of debris on the horizontal upper surface of the plenum only.

4.5.3.2 CRDM Leadscrew and Filter Material

Experimental evidence exists which supports the contention that any TMI-2 core debris which exists on the reactor plenum cover is not pyrophoric. Analyses have been performed on samples of core debris which collected outside of the vessel on filters in the reactor water letdown-purification system. These analyses showed that the filter debris consisted of small particles from <1 to 50 μm, and with a mean particle size of ~6μm. The particles were found to be composed primarily of non-fuel rod components. Over 50 percent of the particles contained stainless steel, inconel, and Ag-In-Cd control material constituents. Based on a limited sampling, most of the Zr-bearing particles were reaction products with uranium, control materials, or structural materials. Thus, the Zr-bearing particles are alloyed as well as physically mixed with other particles, thereby reducing any potential pyrophoricity. Furthermore, although all of the Zr compounds in the debris have not been identified, ZrO_2 has been confirmed. This supports the contention that the zirconium present in the debris has undergone oxidation and reaction with other materials and is therefore, not pyrophoric.

A small sample (estimated to be 20-50 mg) of particulate debris was obtained during the removal of one of the TMI-2 leadscrews and was examined by the licensee. While the exact origin of this sample is uncertain, it showed no pyrophoric reaction upon air drying. It was statically charged; the static charge did not cause the particles to react. A sample of the shavings generated during the cutting of the leadscrew was obtained. The cut was made at an elevation which corresponds to a position on the leadscrew near the plenum cover. Approximately 100 mg of the cuttings containing leadscrew debris were air dried, heated on a hot plate, struck with an electric spark, and heated directly with a flame. No pyrophoricity was indicated; indeed no burning, smoking, or any indication of pyrophoricity was observed. While the quantity of leadscrew debris in the sample was small, it represents a sample where new surfaces have been exposed and thus should be highly reactive. In addition, the cutting operations showed no signs of a pyrophoric reaction nor has long term storage in the containment building of the extracted leadscrews.

A sample of the debris from a 12-inch section of the H-8 leadscrew was sent to a private laboratory for detailed analyses. Preliminary results confirmed the presence of significant quantities of Zr in the debris along with Fe (the major component), U, Te, Cu, and Ni. The principal form of Zr was identified as an intermetallic oxide of the form $FeZrO_4$.

One 9" section of the leadscrew was sent to a second private laboratory for detailed analyses. Preliminary results of these tests and analyses indicate the presence of very little unalloyed Zr. The analysis of this section of leadscrew is specifically directed toward the detection and characterization of any pyrophoric material. In the particles analyzed to date, the Zr exists principally in an alloyed form with Ag, U, or Fe. The presence of ZrO_2 particles has been confirmed. Neither the free metal form nor zirconium hydride have been identified in the particles.

In summary, the analyses performed on the filter debris and leadscrew samples appear to confirm that the TMI-2 particulate debris is not pyrophoric.

4.5.3.3 Zirconium oxidation

The concern over potential pyrophoric materials in TMI-2 is presently focused on the possibility of metallic zircaloy and zirconium hydride fines existing on the horizontal surface of the reactor plenum cover. The manner in which the fuel deteriorates during a severe accident makes the presence of these species in a pyrophoric form highly unlikely. Zircaloy, being a ductile metal even after irradiation, does not break up into small particles under the high temperature steam environment of a LWR accident. Rather, the material oxidizes, and it is the oxide which breaks up as a consequence of thermal shock or abrasion.

Zircaloy which is not substantially oxidized (either to ZrO_2 or to the oxygen-stabilized alpha phase), retains most of its ductility. Therefore, the Zr-bearing particles which might be expected to be carried to the plenum cover would be largely converted to oxide and therefore not pyrophoric. Larger particles from reactor accident experiments (> few mm) are sometimes seen metallographically to be only partially oxidized. A zircaloy metal zone is surrounded by layers of an oxygen-stabilized alpha phase and ZrO_2. In principal, such particles could be pyrophoric if they were broken and a fresh metal surface was exposed to the air. However, particles that large seem to be able to dissipate heat, so that they merely oxidize when the fresh metal is exposed to air, not burn. Experience from the postirradiation examination of kilogram quantities of fuel debris from LWR accident examinations indicate that Zr-bearing particles can be collected, handled, sieved, weighed, etc., in both the wet and dry condition without producing any sustained pyrophoric reactions.

4.5.4 Precautions for the Underhead Characterizaton Program

Based on foregoing information, it is concluded that a pyrophoric event is highly unlikely during the underhead characterization program. However, the following precautions will be implemented to minimize the potential for a pyrophoric event.

a. With the reactor vessel water level above the plenum cover, a video inspection will be performed through the manipulator support tube to better determine the quantity of core debris which may be on the cover.

b. After the video inspection, samples of the observed debris on the cover will be obtained, provided sufficient debris is available. At least two samples will be obtained. (This may be one sample split into several portions.) One sample will be subjected to an immediate test to ascertain if it will burn as was done with the leadscrew sample. The other sample will be analyzed for archive purposes in a laboratory to determine its physical and chemical properties. If insufficient debris is available to obtain a sample, the possibility of a pyrophoric reaction is negated.

4.5.5 Conclusion

It is concluded that the possibility of a pyrophoric event due to the presence of zirconium bearing material is highly unlikely. However, the precautions to be put in place reflect sufficient prudence to permit proceeding with the underhead characterization program.

4.6 HYDROGEN EVOLUTION

As documented in Reference 6, analyses of RCS liquid and RCS high point gas samples indicate that RCS hydrogen evolution has not produced combustible gas mixtures in the RCS high points. Expected RCS hydrogen evolution rates during underhead characterization activities will remain below the level required to produce combustible gas mixtures in RCS high points.

Calculations of the potential flow of gas up the CRDM with the manipulator tube installed result in an exchange rate of more than 5500 cubic feet per day. This corresponds to a free volume turnover of the gas space in the reactor vessel head of approximately seven (7) volume changes per day. Therefore, it may be concluded that with one CRDM manipulator tube open to the building atmosphere and remaining CRDM closures inverted in their respective drives, sufficient air circulation will exist under the reactor vessel head to assure that no hydrogen accumulation will occur.

5.0 10 CFR 50.59 EVALUATION FOR UNDERHEAD RADIATION CHARACTERIZATION

Changes, Test, and Experiments, 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 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

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 following paragraphs are the results of the 50.59 review that was performed for the underhead radiation characterization activities.

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 changes in RCS water level 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 underhead characterization tasks will not create the possibility of an accident or malfunction of a different type than any evaluated previously. This is based on the accident being identified as the inadvertent dropping of an instrument onto the rubble bed in the core. As stated in Section 4.3.1 this is enveloped by previous analysis.

The tasks 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 technical specification limits.

Therefore, it is concluded that performance of underhead radiation characterization does not involve an unreviewed safety question as defined in 10 CFR Part 50, paragraph 50.59.

6.0 CONCLUSION

Based on the radiological and safety evaluations contained in this report, the following articles have been demonstrated:

- o There is no need for maintaining the capability to repressurize the RCS.
- o Load movements have been evaluated and determined to be safe.
- o Occupational exposures will be maintained ALARA.
- o Adequate decay heat removal capability will be maintained.
- o The consequences of potential disturbances of the core are bounded by previous analyses and are acceptable.
- o Adequate measures for the prevention of an RCS boron dilution event will be taken.
- o Release of radioactivity to the environment will be within technical specification limits.
- o Pyrophoricity does not present an undue hazard.
- o Hydrogen evolution will be acceptably low.

In conclusion, the activities comprising radiation characterization under the reactor vessel head may be accomplished without presenting undue risk to the health and safety of the public.

7.0 REFERENCES

1. GPUN letter 4400-82-L-0110, dated July 6, 1982 submitting the Quick Look SER.

2. Addendum to the TMI-2 Decay Heat Removal Report of April 1982, Rev. 1 dated December 1982, prepared by G. A. Hipp, L. L. Losh, and E. R. Miller of Babcock & Wilcox for GPU Nuclear Corporation.

3. GPUN letter 4410-82-L-0073, dated December 30, 1982 regarding incore thermocouples.

4. "Radiological and Safety Evaluation of Ongoing Containment Building Decontamination Activities for TMI-2 Recovery, "Rev. 0, submitted to the NRC by GPUN letter 4410-82-L-0007, Kanga to Barrett, September 23, 1982.

5. NRC letter of September 24, 1982, Snyder to Kanga, regarding releases of radioactivity to the environment.

6. GPUN letter 4410-82-L-0042 dated November 8, 1982, to NRC regarding hydrogen evolution.