Nuclear

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July 6, 1982
4400-82-L-0110

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
CRDM - Quick Look Safety Evaluation Report

Attached, per our commitment in the Recovery Operations Plan Change Request No. 15, is a copy of the Safety Evaluation Report (SER) for the CRDM Quick Look camera insertion, currently scheduled for July 19, 1982. This SER is forwarded to you to provide information related to the planned test. The SER is also provided for your use in reviewing the procedures associated with the proposed test pursuant to Technical Specification 6.8.2. Please note that this SER includes a criticality analysis. This criticality analysis is provided to supplement the criticality analysis provided previously for the APSR insertion test. In addition, your letter of May 12, 1982, approving Recovery Operations Plan Change No. 12 expressed concern about lowering RCS dissolved hydrogen levels to below 5cc/kg. The latest (June 28, 1982) RCS sample shows dissolved hydrogen at 2 cc/kg. This is typical of the latest values. The question of combustible gas is addressed in Section 4.3.1.5 of the SER. Your timely consideration of this document to support the current schedule would be greatly appreciated.

If you have any questions or desire further information, please feel free to contact me.

Sincerely,

J. J. Barton
Acting Director, TMI-2

D009

JJB/RBS/jep

Attachment

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

Safety Evaluation

for

Insertion of a Camera

Into the Reactor Vessel

Through a Leadscrew Opening

Revision 2
July 1982

1.0 TITLE:

Safety Evaluation For The Insertion Of A Camera Into The Reactor Vessel Through A Leadscrew Opening

2.0 PURPOSE:

The insertion of a camera into the reactor vessel through a leadscrew opening, the "Quick Look," will provide the first visual assessment of conditions inside the reactor vessel. CCTV camera inspections will include the following:

o At the location of removed leadscrew

- guide tube brazements
- guide tube C-tubes
- control rod spider hub
- control rod pins
- fuel assembly end fitting
- end fitting guide tabs

o Adjacent fuel assembly locations

- fuel assembly end fitting
- end fitting guide tabs

o If there is no end fitting present

- condition of the core and debris over as broad an area as possible

The Quick Look has the following suggested data acquisition goals:

o Sample gas vented from CRDM motor tube during venting operations

o Sample reactor coolant drained from motor tube during venting operations

o Monitor and record weight indications during uncoupling operations

o Swipe samples from various parts of the leadscrew

o Retrieve a representative water sample from the reactor vessel plenum and/or fuel region

Changes in the reactor coolant system (RCS) operating parameters required for the Quick Look are 1) reduced RCS pressure, 2) venting of the RCS, and 3) lowered RCS water level. These changes result in breaching of the reactor coolant pressure boundary (RCPB) and reduced decay heat removal capabilities.

To complete the safety evaluation, the following concerns associated with the Quick Look have been evaluated:

- the potential releases of radioactivity to the containment
- the potential releases of radioactivity to the environment
- the effects on reactivity as a result of potential disturbances of the fuel
- the effects of RCS draindown on decay heat removal capabilities
- the potential for inadvertent boron dilution
- the release of gases from the RCS to the containment atmosphere
- occupational exposure, and
- the effects on RCS chemistry as a result of breaching the RCPB

3.0 SYSTEMS AFFECTED:

3.1 The Quick Look will involve the following systems:

- Reactor Coolant System
- Feedwater System
- Reactor Coolant System Level Monitoring
- Standby Reactor Coolant Pressure Control System
- Mini Decay Heat Removal System
- Letdown System
- Nitrogen System

3.2 Drawings

Flow Diagram - Reactor Coolant, Makeup & Purification - Burns & Roe Drawing 2024

Flow Diagram - Feedwater & Condensate - Burns & Roe Drawing 2005

Flow Diagram - Standby RC Pressure Control System - Burns & Roe Drawing M022

Flow Diagram - Nitrogen for Nuclear and Radwaste Systems - Burns & Roe Drawing 2036

3.3 Documents

3.3.1 The following sections of the TMI-2 FSAR describe the systems involved in the Quick Look.

Reactor Coolant System - Sections 4.0 & 5.0

Feedwater and Condensate - Section 10.4.7

Makeup and Purification (Letdown) - Section 9.3.4

3.3.2 Technical Specifications

The following technical specifications govern the systems listed in Section 3.1.

Boron Control 3.1.1 & 4.1.1
Reactor Coolant System 3.4 & 4.4

3.3.3 System Descriptions

3.3.3.1 Standby Reactor Coolant Pressure Control System - refer to the Standby Reactor Coolant Pressure Control System - System Description

3.3.3.2 Mini Decay Heat Removal System - refer to the Mini Decay Heat Removal System - System Description

3.3.3.3 RCS Level Monitoring System - System Description

3.3.3.3.1 Design Bases

The RCS level monitoring system will perform the following functions.

1) Provide the control room operators with direct indication of RCS water level during the process of lowering the RCS water level.

2) Provide RCS water level indication between the elevation of the RCS high points (pressurizer, top of the hot legs) and the elevation of the reactor vessel hot leg nozzles. Accuracy of indication will be sufficient to allow personnel to reliably determine when the RCS water level is at the elevation required for the camera inspection through a leadscrew opening.

3) Compensate for the RCS nitrogen over-pressure.

4) Compensate for any differential pressure that may be established between the fuel handling building (instrument location) and the reactor building.

3.3.3.3.2 System Design, Installation and Operation

The RCS water level monitoring system will be designed and installed as an "Important to Safety" (ITS) system.

The RCS water level monitoring system uses a level transmitter connected to the normal decay heat removal line. The transmitter, located in the fuel handling building, will sense the head of water in the RCS above the centerline of the reactor vessel hot leg nozzle and transmit an electrical signal to a digital indicator and strip chart recorder located in the main control room. It will be calibrated to measure the column of water between the reactor vessel hot leg nozzle centerline and the top of the hot leg. The indicated value will be actual water level since design features will compensate the reading for the RCS nitrogen overpressure and differential pressure between fuel handling and reactor buildings.

The range of indication for this system will be 0 to 600 inches of water with a system accuracy of ± 5 inches of water.

A local level indicator in the fuel handling building is also provided and connected in parallel with the level transmitter. Its range is also 0 to 600 inches of water with an accuracy of ± 6 inches of water.

This is the system that will be used to determine when the level in the RCS is within the range of elevations for camera insertion through the leadscrew opening.

The level transmitter and indicator will be installed with provisions for isolation, calibration, and maintenance. Piping and tubing will be leak-tested in accordance with ANSI B31.1 1980.

4.0 EFFECTS ON SAFETY

4.1 Safety functions of the systems listed in Section 3.1 are described in the documents listed in Section 3.3.

4.2 Safety Functions

4.2.1 The safety functions of the RCS are to contain the radioactive reactor coolant and to remove reactor decay heat. Containment boundary isolation is the only safety function of the nitrogen and feedwater systems in the present operating mode. The letdown and RCS level monitoring systems do not perform any safety functions for the Quick Look.

4.2.2 The functions of the Standby Reactor Coolant Pressure Control System are to:

a. Maintain the Reactor Coolant System in a water-solid condition for long term reactor core natural circulation cooling by automatically compensating for coolant volumetric decreases in the Reactor Coolant System due to system leakage or thermal contraction.

b. Maintain the Reactor Coolant System pressure at a preset pressure with the pressurizer solid and pressurizer heaters out of service.

c. Provide adequate NPSH to the reactor coolant pumps if it becomes necessary to use one.

d. Provide capability for makeup water to the RCS when the RCS is in a drained condition.

4.2.3 The functions of the Mini Decay Heat Removal System (MDHR) are as follows:

a. Remove heat from the reactor coolant system by forced circulation through the core.

b. Provide a method of removing heat from the reactor coolant system during reactor vessel head removal and vessel defueling.

c. Provide piping connections for future cleanup of the reactor coolant system.

d. Provide a means of sampling the reactor coolant system utilizing the Mini Decay Heat Removal System.

e. Provide a means of backup pressure control for the Reactor Coolant System.

The MDHR only serves as a backup to the loss to ambient cooling mode.

4.3 Quick Look Affects On Safety Functions

4.3.1 System Performance

4.3.1.1 Environmental Releases

The Quick Look may result in an increase in radiological environmental releases due to the release of Kr-85 from the RCS. This potential release has been evaluated using the following assumptions: 1) the entire RCS inventory of dissolved and free Kr-85 is released in one hour into the containment purge exhaust, 2) the Kr-85 is diluted by a plant vent stack flow rate of 100,000 cfm and, 3) no credit is taken for Kr-85 dilution in the containment.

The total quantity of Kr-85 in the RCS available for release was calculated to be approximately 30 Ci. This is based on the following:

- o It is assumed that Henry's Law for dissolved gases vs. free gases applies.
- o The RCS vapor space is conservatively assumed to be 190 ft^3.
- o RCS Kr-85 activity is 0.07 µCi/cc (per 7/20/81 sample results).
- o No processing of RCS has taken place.

Using the calculated release of 30 Ci and the guidance provided in Regulatory Guide 1.109, the increased dose at the nearest residence was calculated to be 2.1×10^{-5} mrem (total body dose).

Actual releases from the plant vent stack, including during times of containment purging, have been a small fraction of the limits specified in the TMI-2 Technical Specifications. With the exception of RCS venting, the Quick Look is not expected to result in an increase in releases of radioactivity to the environment. All releases from the containment as a result of the Quick Look will be within the limits of the TMI-2 Technical Specifications.

4.3.1.2 Reactivity Changes

Reactivity changes as a result of postulated fuel disturbances or changes are described in Appendix A. Appendix A has been prepared by Babcock and Wilcox and is entitled "Methods and Procedures of Analysis for TMI-2 Criticality Calculations to Support

Recovery Activities Through Head Removal." This report shows that the TMI-2 reactor will be maintained subcritical during the activities associated with the Quick Look considering any credible fuel configuration inside or outside of the core region and considering the effects of postulated fuel disturbences or changes in physicel conditions. This assurance will be provided at all times when the reactor coolant is borated to a concentration of 3500 ppm or greater.

The water level in the secondary side of the steam generators will be maintained at an elevation below that of the RCS water level. This will ensure that should there be an accumulation of fuel in the steam generator tubes, the accumulation of fuel will remain subcritical at all times during the activities associated with the Quick Look.

4.3.1.3 Decay Heat Removal

An analysis of the effects of the draindown of the RCS on decay heat removal capabilities has been performed. The analysis is described in Appendix B. The results of the analysis show that:

a) There appears to be some migration of water through the loops with the result that the current heat rejection mode includes system components beyond the reactor vessel and head.

b) Lowering the reactor water level and isolating the system components may cause an increase in reactor water equilibrium temperature.

c) The reactor temperature increase to achieve equilibrium conditions depends upon the heat sink temperature (i.e., containment temperature).

d) The anticipated heatup rate, after draindown to elevation 323'-6", is expected to be less than 5 F per day in mid 1982 at 100 F containment temperature. Thus, ample time is available to monitor actual heatup rate and determine if enhanced heat removal is required to maintain equilibrium peak temperatures below approximately 190 F.

e) The enhanced heat removal, if required, can be accommodated by existing systems such as:

- feed and bleed through letdown and the standby pressure control system

- refill of the Reactor Coolant System

- Mini Decay Heat Removal System

This analysis is based on an RCS water level one foot above the plenum cover (elevation of the plenum cover is 322'-6"). For the Quick Look, the RCS water level will be lowered to between elevations 331' and 335'. The heat up rate discussed in item d, above, should be less for the Quick Look because of the additional water in the RCS for the Quick Look.

4.3.1.4 Boron Dilution

The potential for a boron dilution event has been considered and potential dilution paths have been identified. Appropriate administrative procedures and precautions have been developed to preclude a boron dilution event. Appendix C describes the actions taken to prevent boron dilution as well as a discussion of the actions which will be taken in the unlikely event that a dilution event occurs. The actions to prevent boron dilution include 1) removing spool pieces to assure system flowpath isolation, 2) draining systems to reduce pressure differentials with the RCS and, 3) valve line-ups that provide a greater margin of protection against leakage into the RCS.

A boron dilution event caused by a secondary to primary leak through steam generator tubes will be prevented by 1) maintaining secondary water level below RCS water level in the steam generators and, 2) maintaining equal N_2 pressures on the primary and secondary sides of the steam generators.

Throughout the Quick Look, the capability to borate the RCS will be maintained in accordance with the TMI-2 Technical Specifications.

4.3.1.5 Gas Releases

The venting of the RCS will result in the release of trapped gases including Kr-85 and hydrogen to the containment atmosphere. The consequences of releasing the Kr-85 first to the containment and then to the environment are discussed in Section 4.3.1.1. The consequences of releasing the hydrogen are evaluated below.

The total quantity of hydrogen in the RCS was estimated based on the following assumptions:

- o RCS has not been processed.
- o Volume of free gas in the RCS was determined by a test which conservatively ignored differences in water head.
- o Ratio of hydrogen to nitrogen in the free gas is the same as the ratio in the RCS samples.
- o Henry's Law for dissolved gases vs. free gases applies.

Since no credit is taken for RCS processing, the percentage of hydrogen in the free gas should be consistent with the percentage of hydrogen in the RCS sample results. The RCS sample results indicate approximately 7cc/Kg hydrogen and 9cc/Kg nitrogen. The lb - moles of hydrogen and nitrogen in the free gas required to attain the relative sample concentrations was determined using Henry's Law. It was found that approximately 40 percent of the free gas will be hydrogen and the remaining 60 percent will be nitrogen.

The amount of free gas in the system was determined by using the following equations:

Equation 1 $$n = \frac{(P_1\ P_2)\quad(\Delta V)}{(\Delta P)\ R\ T}$$

Where: P_1 = initial pressure of system in psia

P_2 = final pressure of system in psia

ΔP = change in system pressure in psia

ΔV = change in system water volume in ft^3

T = absolute temperature of system in °R

R = universal gas constant =

$$10.729 \frac{Ft^3 - lb}{lb\text{-}moles - {}^\circ R - in^2}$$

n = lb-moles of free gas

Equation 2 $$V = \frac{n\,R\,T}{P}$$

Where: n = lb-moles of free gas from eq 1

R = same as Equation 1

T = 530°R (i.e. 70°F)

P = 14.7 psia

V = standard cubic feet

No credit was taken from the difference in water head between the pressure sensing point and the possible location of the free gas. This assures a conservative estimate for the free gas in the RCS. Using the results of the RCS test, it was determined that approximately 370 scf of free gas is in the RCS. Assuming average conditions in the RCS of 90 F and 100 psig, this is equivalent to approximately 48 ft^3 of void space.

The amount of free gas that is hydrogen would be about 145 scf. Such a high concentration of hydrogen indicates that the RCS water contains approximately 700 to 800 additional standard cubic feet of hydrogen. Therefore the total quantity of hydrogen in the RCS was conservatively estimated at 1000 scf.

The gas vented from the RCS is assumed to be 100 percent hydrogen and will be discharged into a dilution flow stream with a minimum dilution factor of 25. This will prevent the discharge of a flammable hydrogen mixture to the containment atmosphere. The discharge will be directed away from where personnel are expected to be.

During leadscrew withdrawal and cutting operations the RCS will be at atmospheric pressure. The RCS surface area exposed to the atmosphere via the open CRDM motor tube will be less than 4 in^2. Hydrogen offgassing via this small surface area has been calculated to be approximately 0.03 scfm. Such a low release rate should not present a hydrogen flammability hazard when vented directly to the containment.

4.3.1.6 Occupational Exposures

The total exposure for the Quick Look is estimated to be 50 to 150 man-rem. The activities included in this estimate are those associated with:

- o decontamination and contamination control
- o rigging
- o reactor coolant system venting
- o installation of temporary power and lighting
- o removal of a leadscrew and inspection inside the reactor vessel

It was estimated that approximately 300 in-containment man-hours are associated with the covered activities. These man-hours will be spent in radiation fields that vary from 0.15 R/hr to 30 R/hr. The calculated man-rem for all activities was 100 man-rem. Because of the uncertainty in the dose rates and man-hours, the man-rem for the activities may vary by ± 50%. Therefore, the total exposure for the Quick Look is estimated to be 50 to 150 man-rem.

The quantity of Kr-85 contained within the RCS and available for release to the containment during RCS venting has been estimated to be up to 30 curies (see Section 4.3.1.1). The estimate was based on RCS sample data prior to RCS processing. Of the estimated total Kr-85 inventory approximately 7 curies are estimated in the free gas volume. In order to minimize the occupational exposure from Kr-85 during the venting operation the following requirements will be factored into the RCS venting procedure. The discharge point will be at least ten feet from personnel and directed away from personnel. External exposures to personnel will be monitored by personnel dosimetry. In addition, air sampling will be performed during venting and samples analyzed for Kr-85, particulates, and tritium.

The release of the Kr-85 to the containment may require continuous purging of the containment for a period of time after the venting operation in order to reduce the Kr-85 concentration in the containment atmosphere to as low as reasonably achievable. The exact length of time that purging will be required depends on the amount of Kr-85 released to the containment, but should be less than 24 hours based on an assumption that 10 equivalent air changes will be more than sufficient for flushing the containment atmosphere.

During RCS venting operations there is a possibility of releasing additional particulates and tritium to the containment atmosphere. The level of contamination of the RCS water is currently less than the contamination level of the water which was removed from the containment sump. Therefore no significant increase in containment particulate and tritium airborne contamination is expected during RCS venting.

4.3.1.7 RCS Chemistry

Recovery Operations Plan Change No. 12 issued May 12, 1982 by the NRC approves GPUN's Recovery Operations Plan Change Request No. 14 on Reactor Coolant System Chemistry Specifications. This change was necessary for RCS processing. The effects on RCS chemistry created by breaching the RCS boundary for the Quick Look are bounded by the effects created by RCS processing.

4.3.2 Quality standards: The GPUN QA Plan will be implemented as applicable.

4.3.3 Natural phenomena protection: Not applicable

4.3.4 Fire Protection

Fire protection for the Quick Look program will be provided in accordance with the requirements of the TMI-2 Fire Hazards Analysis Report and TMI-2 Administrative Procedure 1034, Control of Combustible Materials.

The estimated increase in containment combustible loading for the Quick Look is 0.022 lbs (wood equivalent)/ft^2. Based on this small increase in combustible loading, additional fire protection is not required for the Quick Look. At the conclusion of the Quick Look, all Quick Look materials that are not removed from the containment will be factored into the TMI-2 Fire Hazards Analysis.

4.3.5 Environmental Qualification: Not applicable.

4.3.. Missile protection: Not applicable.

4.3.7 High energy line pipe breaks: Not applicable.

4.3.8 Electrical separation: Not applicable

4.3.9 Single failure criteria: Discussed in Appendix C in conjunction with the potential for boron dilution of the RCS.

4.3.10 Separation criteria: Not applicable.

4.3.11 Containment isolation: Not applicable.

4.3.12 Materials compatibility: All materials which may contact reactor coolant will be fabricated of materials compatible with RCS water chemistry.

4.4 Determine if the change, test, or experiment will increase the probability of occurrence or the consequences of an accident and state the basis for that determination:

The activities associated with the Quick Look will not increase the probability of occurrence or the consequences of an accident. This is based on the effects on reactivity changes discussed in Section 4.3.1.2 and Appendix A. It is also based on the precautions and measures that will be taken to prevent a boron dilution event (Section 4.3.1.4) and to prevent a concentration of flammable gases (Section 4.3.1.5).

4.5 Determine if the change, test, or experiment will increase the probability of a malfunction of equipment ITS and state the basis for the determination:

The activities associated with the Quick Look will not increase the probability of a malfunction of equipment ITS. This is based on equipment ITS being operated in a manner for which it was designed. In addition, the Quick Look activities will not adversely impact any equipment ITS. Where there are design changes to existing equipment ITS, the design change will satisfy the ITS requirements commensurate with the importance to safety of the item during the recovery period.

4.6 Determine if the proposed change, test, or experiment creates a possibility for an accident or malfunction of a different type than any previously identified in the SAR and state the basis for that determination:

The activities associated with the Quick Look do not create a possibility for an accident or malfunction of a different type than any previously identified in the SAR or other safety evaluations. The effects on reactivity changes have been shown not to constitute a safety concern in the Safety Evaluation for the Axial Power Shaping Rod (APSR) Insertion Test, and as discussed in Section 4.3.1.2 and Appendix A. The APSR test Safety Evaluation was previously reviewed and approved by NRC. The other areas where this could have occurred, boron dilution and concentration of flammable gas, have been eliminated as discussed in Sections 4.3.1.4 and 4.3.1.5.

4.7 Determine if the proposed change, test, or experiment decreases the margin of safety as defined in the basis of any technical specification and state the basis of that determination:

The activities associated with the Quick Look will not reduce the margin of safety as defined in the basis of any technical specification. With the exception of Standby Pressure Control System, the Quick Look activities will not modify the operation of existing systems.

4.8 Determine if the proposed change, test, or experiment will violate any technical specifications and state the basis for that determination:

The activities associated with the Quick Look will not violate any technical specifications based on a review of the TMI-2 Technical Specifications.

4.9 Determine if the proposed change, test, or experiment will violate any license requirements or regulations and state the basis for that determination:

The activities associated with the Quick Look will not violate any license requirements or regulations. This is based on a review of the TMI-2 Technical Specifications and the TMI-2 FSAR.

4.10 Determine if the proposed change, test, or experiment involves a radiological safety concern and state the basis for that determination:

The activities associated with the Quick Look do not involve an unacceptable radiological safety concern. The basis for this is that the releases of radioactivity to the environment, including the contribution due to venting of the RCS, will be well within the limits of the TMI-2 Environmental Technical Specifications, as discussed in Section 4.3.1.1. As described in Section 4.3.1.6, the activities associated with the Quick Look do not involve unacceptable radiological safety concerns to the workers.

5.0 Conclusion

The activities associated with the Quick Look do not constitute an unreviewed safety question. These activities can be performed within existing TMI-2 Technical Specifications and present no undue risk to the public health and safety.

BAW-1738

June 1982

APPENDIX A

METHODS AND PROCEDURES OF ANALYSIS
FOR
TMI-2 CRITICALITY CALCULATIONS TO SUPPORT
RECOVERY ACTIVITIES THROUGH HEAD REMOVAL

Babcock & Wilcox

a McDermott company

NOTE

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BAW-1738

June 1982

METHODS AND PROCEDURES OF ANALYSIS
FOR
TMI-2 CRITICALITY CALCULATIONS TO SUPPORT
RECOVERY ACTIVITIES THROUGH HEAD REMOVAL

Prepared by

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1.0 INTRODUCTION AND SUMMARY

1.1 Purpose and Scope

This report was prepared to document criticality safety results obtained from the analyses of various geometrical configurations of moderator, reflector, and fuel. These configurations represent both credible and hypothetical fuel arrangements which could now exist or could occur in the Reactor Coolant System (RCS) as a result of activities relating to thru-head inspections and reactor vessel closure head removal.

Examples of some of the proposed activities that could produce fuel rearrangements or otherwise affect the subcriticality of the fuel system are:

- Insertion of the Axial Power Shaping Rods (APSRs)
- Control Rod Drive Mechanism (CRDM) Uncoupling Attempts
- Insertion of Inspection and Sampling Equipment into the Reactor Vessel through Penetrations in the Head
- Removal of the Reactor Vessel Head

The purpose of the analytical assessment is to demonstrate that during all of these proposed activities the TMI-2 reactor will be maintained in a safe shutdown condition at all times. The specific objectives of this report are:

- To evaluate the reactivity of postulated TMI-2 core configurations.
- To evaluate the reactivity of potential fuel accumulations outside the core region
- To evaluate the potential reactivity effects of various perturbations resulting from the proposed activities.

- To verify that a boron concentration of 3500 ppm will maintain an adequate margin of subcriticality under all postulated credible conditions.

1.2 Report Organization

Section 2.0 provides a summary description of the various computer codes that are used in reactor and nuclear criticality safety applications. These are the codes that were used by B&W in the criticality studies to support the proposed activites (see Section 1.1) and in the various studies performed immediately following the accident. Section 3.0 describes the analyses that were performed to assess the potential in-core reactivity effects of the proposed activities. Section 4.0 describes the similar analyses that were performed to assess the potential reactivity effects of fuel movement out of the core that may result from these activities. Section 5.0 presents an overall summary of the results and conclusions of this report and identifies the conservatisms and calculational uncertainties inherent to the fuel models.

1.3 General Introduction

Evaluations of core subcriticality and loose fuel configurations with high reactivity potential were begun soon after the TMI-2 incident. The purpose of those calculations was to evaluate the reactivity of the core considering a wide range of core damage possibilities, and to provide a basis for selecting a safe RCS boron concentration.

A general, brief description of the original core is as follows. The core region without fuel is an approximate cylinder 5.4 feet in radius and 13.8 feet high, filled with water. The water may contain other chemicals such as boric acid and lithium hydroxide. The fuel was in the form of 177 fuel assemblies, divided into 3 batches of 61, 60 and 56 assemblies. The assemblies within each batch were manufactured identically except for the difference in the U-235 enrichment. The assemblies are designed with a 15 by 15 array of cells on a square pitch. Each assembly contains 208 fuel rods with the 17 other cells consisting of one central instrument tube and 16 guide tubes. The guide tubes may contain either burnable poison rods, control rods, axial power shaping rods or water. The fuel rods contain stacked UO_2 pellets with zircaloy cladding and are surrounded by the water moderator. The cells within the assembly are held in a square array by Inconel grids.

Following cooldown of TMI-2, plans were made for decontamination of the containment building and eventual removal of the fuel from the damaged reactor. A prerequisite step for fuel removal is performance of remote inspection of the interior of the reactor vessel, followed by removal of the reactor vessel closure head. Since these activities may involve some perturbation of the fuel, the reactivity affect of such activities must be examined. The primary safety goal is to ensure a degree of subcriticality which ensures shutdown margin both in the core or in postulated fuel accumulations outside the core.

Since the shutdown of the reactor in 1979, gradual changes have occurred in the core and the coolant. These changes include a reduction in the primary coolant and fuel pressures and temperatures, and a change in the core fission product inventory due to radioactive decay. These changes were expected to have little effect on the long term reactivity status of the core. Nevertheless, the results of previous criticality analyses were reviewed and a reevaluation of those analyses has been included in this report to form a basis for further investigations of criticality safety.

Other changes in the core or fuel configurations outside the core could conceivably occur as a result of the activities to be undertaken on the reactor head. The extent of any such changes is dictated by the condition of the damaged core (see Figure 1-1 for possible regions of damage). If the core damage is slight, the activities proposed through head removal activities should do little more than disturb some of the damaged fuel rods or fuel pellets. Conversely, if the existing damage to the fuel is extensive, any disturbance of core components may initiate a greater degree of fuel rearrangement. This rearrangement could be the result of fuel assembly collapse due to the driving force on the APSRs, impact of a dropped fuel assembly end fitting, or impact from the release of an unsupported leadscrew during CRDM uncoupling operations. Repositioning of the APSRs could also introduce direct reactivity changes.

During the insertion of inspection and sampling equipment through the reactor vessel closure head penetrations, there is the possibility of dropping equipment or dislodging damaged material. Either action could cause some amount of fuel to be repositioned.

Since the activities relating to closure head removal could conceivably caus. fuel to drop from the core region to the reactor vessel lower plenum, additional analyses were required to evaluate criticality safety for fuel accumulations in that region. A review of fuel accumulation areas outside the core region showed that the reactor vessel lower plenum was the most credible and also potentially the most reactive of the potential fuel collection areas. Section 4.0 describes the analyses performed for the reactor vessel lower plenum.

1.4 Safety and Licensing Criteria

A shutdown margin of at least 1% $\Delta\rho$ ($K_{eff} < .99$) has been used in determining the safety of the analyzed fuel configurations and boron concentrations. This is the criterion that was applied during the reactor cooldown period and is defined in the Standard Technical Specification[1] as applicable to the shutdown mode of reactor operation.

TMI-2 was officially placed in the shutdown cooling mode on February 13, 1980 by the current Technical Specification[2] for that plant. The technical specifications also define an additional shutdown mode called the recovery mode in which the reactor is subcritical with an average coolant temperature less than 280°F. The technical specifications also define the limiting conditions for the moderator boron concentration and minimum coolant temperature.

This mode applies during the long term cooling of the core including facility cleanup and recovery operations.[3] Therefore, the specific criticality safety requirements for the activities proposed through reactor vessel head removal pursuant to the technical specifications, are as follows:

Criticality Safety Requirements

	Requirement
Subcritical multiplication, K_{eff}	Less than .99
Coolant boron concentration	Greater than 3000 ppm
	Less than 4500 ppm*
Coolant Temperature	Greater than 50° F
	Less than 280° F

*Less than 4500 ppm is not specifically a criticality requirement but was included in the technical specifications to prevent boron precipitation.

In order to provide additional margin of safety, a conservatism of 2% $\Delta\rho$ shutdown margin was used in this work. Calculational biases which resulted in increases to the reactivity of the system have been included in all analytical results reported in Sections 3 and 4 and the effective neutron multiplication of the analytical system is less than 0.98 in all cases.

1.5 Results and Conclusions

For a detailed description of the various analyses performed for the report refer to Sections 3.0 and 4.0.

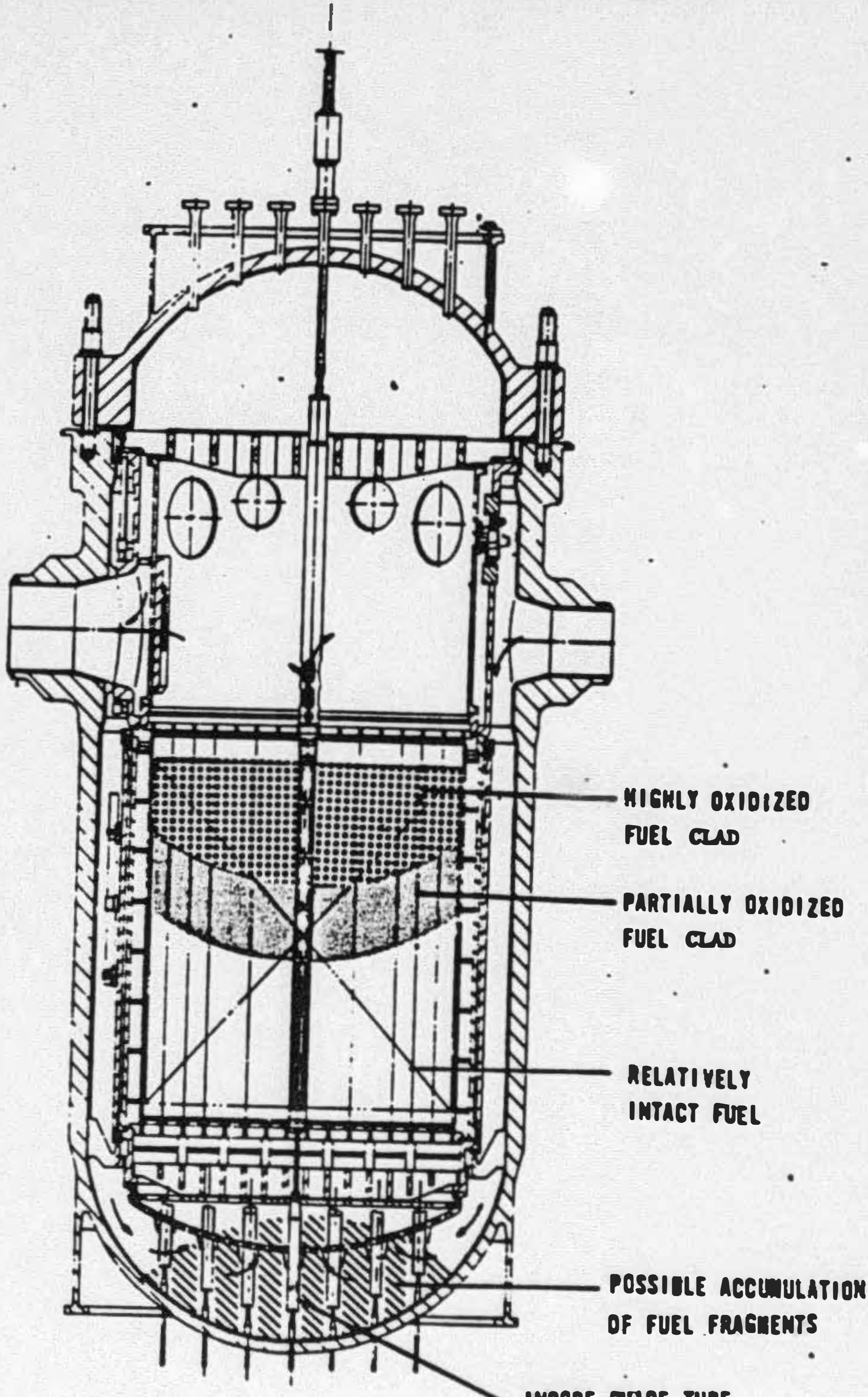
The analyses show a K_{eff} of 0.943 for the core with maximum credible damage (50% damaged and 50% undamaged) at 3500 ppm boron concentration. The K_{eff} for a hypothetical core model (total fuel collapse) is 0.977 with 3500 ppm boron.

The most reactive credible configuration of fuel in the bottom of the reactor vessel has a K_{eff} of 0.979 with 3500 ppm boron.

The preceding values of K_{eff} are based upon standard criticality procedures which have a large degree of conservatism. Further analyses of the more probable K_{eff} of the reactor which reflect the realistic estimate of the reactor's true K_{eff} give a value less than 0.902 for both the core and out of core regions.

The calculations show that the required subcriticality (K_{eff} of .99) is met by a margin of 1% $\Delta\rho$ for both the core and the most reactive fuel region outside the core. Consequently, the TMI-2 reactor will be maintained subcritical during the proposed head removal activities considering any credible fuel configuration inside or outside of the core region and considering the effects of postulated fuel disturbances or changes in physical conditions. This assurance will be provided at all times when the reactor coolant is borated to a concentration of 3500 ppm or greater.

Figure 1-1 POSSIBLE REGIONS OF DAMAGED FUEL IN THE REACTOR VESSEL



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2.0 CRITICALITY METHODS AND CODES

2.1 Introduction

2.1.1 Physics Methods for Criticality Calculations

The most exact description of the time dependent neutron behavior of a particular system is obtained from the solution of the neutron transport equation. If the system contains fuel, such as ^{235}U, then the neutron population will be a function of time and the system will either be subcritical, critical or super-critical. Criticality calculations are only concerned with the system in the sub-critical or critical state. When the system is critical, it is producing as many neutrons as those being lost and the behavior of the neutron population is independent of time. When the system is subcritical, the neutron population will decay with time unless it is sustained by a neutron source. If it is sustained by a source, it will achieve a steady state population level which is independent of time. The degree of subcriticality is determined by the ratio of productions to losses.

When the neutron population is independent of time the most exact method of describing the neutron behavior is the time independent neutron transport equation. This equation considers the neutron behavior to be dependent on six variables, three position (x,y,z) and three velocity (V_x,V_y,V_z). In addition, the neutron transfer probability function is dependent on three velocity variables $(V_x',V_y',V_z' \rightarrow V_x,V_y,V_z)$.

There is no general solution to the transport equation. Therefore, the physics methods employed in the computer codes used to calculate criticality utilize varying degrees of approximations for the solution of the transport equation. While there are some codes that can treat the neutron behavior to a limited degree in all six variables (x,y,z,V_x,V_y,V_z), the standard techniques in criticality calculations are to separate space and velocity. The codes which are used to accurately compute the neutron velocity variables are termed spectrum codes. These codes employ more approximations in their treatment of the spatial variables (x,y,z) than they do for the velocity variables. The spectrum codes are used to furnish velocity weighted cross sections to the spatial codes.

The standard practice in the spectrum codes is to transform the velocity variables to a scaler function of energy, or lethargy and a solid angle. The spatial codes employ these same variables including as many as 100 energy groups or as few as 2 groups to accurately predict the neutron behavior. When the spatial codes explicitly treat the angular direction of the neutrons, they are termed transport theory codes. If the spatial codes approximate the angular direction of the neutrons as isotropic, they are termed diffusion theory codes. The codes for calculating criticality always make some approximation of the angular neutron behavior, but never assume a completely isotropic behavior.

Section 2.2 discusses the codes and their approximations in more detail.

2.1.2 Selection of Techniques for TMI-2

Two classes of criticality techniques have been used in this study, reactor and nuclear. The reactor criticality techniques are those used for reactor design. The nuclear criticality techniques are those used for out of core calculations of spent fuel pools, reprocessing plants, etc.

In a standard PWR core the fuel material is arranged in a regular lattice array of fuel pins surrounded by a water moderator. The calculational tools required to analyze such arrays have been highly refined by benchmarking them against numerous measurements ranging from operating reactors to critical experiments. The fact that the reactor fuel is in a well moderated regular array means that the calculational techniques do not have to be versatile as far as the ability to handle widely different fuel arrangements or spectra is concerned. Rather, these reactor criticality techniques must be very accurate in their predictions of reactivity and neutron reaction rates. These techniques utilize the computer codes NULIF[1] and PDQ[2] for standard calculations, and ANISN[3], DOT[4] and KENO[5] for special cases.

The requirements imposed on the calculational techniques used in nuclear criticality analysis are different from those of reactor criticality analysis. Since nuclear criticality safety is concerned with a wide variety of fuel arrangements and spectra the calculational tools must be very versatile. Precise knowledge of reactivity is secondary in importance to the requirement that the system in question be subcritical by an acceptable limit. The acceptable limit can be adjusted to allow for uncertainties in calculated reactivity without imposing an undue penalty on the processes involved. The nuclear criticality techniques use the computer codes KENO[5], NITAWL[6], XSDRNPM[7,8] and ANISN[3].

Both reactor and nuclear criticality techniques were used for the analyses of the damaged TMI-2 reactor. The core damage studies suggest that part of the core is in the original lattice while other parts of the core have probably been damaged to the extent that the fuel material configuration is unknown. Furthermore, fuel debris may be out of the core region. These unusual conditions require the combined techniques of reactor and nuclear criticality analysis. The numerous cases to determine the most reactive configuration for the damaged fuel were principally based on reactor criticality calculations while the geometrical models were principally based on nuclear criticality calculations. The results of both calculations were cross-checked using both techniques.

2.2 Computer Code Descriptions

Subsection 2.2.1 describes the computer codes used in the reactor criticality calculations and Subsection 2.2.2 describes the codes used in the nuclear criticality calculations.

2.2.1 Reactor Criticality Codes

2.2.1.1 NULIF

The NULIF[1] code serves as the basis for the velocity weighted cross sections used in ANISN[3], DOT[4], PDQ[2] and KENO[5]. The production library for NULIF contains 111 energy group cross sections up to 10 Mev. These cross sections were developed from a velocity weighted master library containing 207 energy groups up to 15 Mev. The master library was developed from velocity weighting of the point, or continuous energy ENDF/B cross sections which extend to 20 Mev.

The evaluation and collection of basic nuclear cross section data has been greatly simplified by the Evaluated Nuclear Data Project at Brookhaven National Laboratory. This project is sponsored by DOE with technical direction provided by the National Neutron Cross Section Center. The objective is to produce the ENDF/B nuclear data library tapes which contain a collection of documented nuclear data evaluations stored with a specified format. These ENDF/B tapes are the primary sources of nuclear cross section data for the B&W Master Library.

The Master Library contains 207 group cross section data for each material and resonance parameters for each resonance including s-wave, p-wave, and unresolved data. For standard core analysis calculations, sufficient accuracy is obtained with the less detailed 111 group production library. In addition, for resonance materials of low concentration, Doppler broadening and self-shielding effects are negligible, and their resonance absorption can be represented by smooth-group cross sections. The p-wave resonance contributions are usually small and can also be represented by smooth-group cross sections as can the unresolved resonances (or high-energy portions of the unresolved resonances) of many isotopes.

The NULIF code computes the neutron spectrum and provides spectrum-weighted cross sections using the Production Library and physical and geometrical descriptions of the fuel cluster, fuel cell, or other material regions. The spectrum is computed by solving the P_1 approximation to the Fourier transform of the space dependent neutron transport equation. Leakage effects are approximated by input of a fundamental mode buckling for the region. Self-shielding effects caused by the heterogenous lattice configuration and the different compositions of unit cells, are taken into account in the spectrum calculation.

The neutron source for the spectrum calculation is either a normalized fission source distribution, or some fixed source with a fission source. Theoretically, the slowing down treatment for hydrogen moderator is exact; the Greuling-Goertzel[9] approximation is used for other materials. For isotopes with neutron resonances in the epithermal range, the effective resonance integral for each resonance peak in each microgroup is computed by Dresner's[10] method using Sauer's[11] model to compute the Dancoff factor for close-packed fuel pin lattices. Both self-shielding and Doppler broadening are taken into account. In computing the epithermal spectrum, both inelastic scattering and (n,2n) reactions are taken into account by the use of downscatter matrices.

Below 1.85 eV, both upscatter and downscatter may occur; this is designated the thermal energy range. Because of the upscatter effects, the ne··tron spectrum in this range is solved by an interative procedure. Since upscattering above 1.85 eV is neglected, the sources for the thermal spectrum calculation are those neutrons scattered from above 1.85 eV into each microgroup below 1.85 eV. In the thermal region, the group structure is sufficiently fine to mock up all the resonances explicitly by group cross-section values.

Heterogenous fuel cell local flux depression effects are obtained by the method of Amouyal, Benoist, and Horowitz[12]. In this manner, flux depression factors for each of the thermal groups are applied to the cross section of each cell region before computation of the thermal flux spectrum. For water-moderated lattices, the scattering matrices are interpolated from temperature dependent scattering law (Haywood Model) data.

If the space dependent fundamental mode buckling is known for a region, the NULIF code will accurately compute the K_{eff} ($K_{eff} = K_\infty e^{-B^2M^2}$)[13] for the system. This is discussed further in Section 2.3 on the benchmarking of the codes.

2.2.1.2 PDQ

Basically, PDQ07[2] is a few-group, diffusion-depletion program used to obtain the group dependent neutron flux and reaction rate distributions (in one-, two-, or three-dimensions) throughout a reactor core during its operating life. The input consists of the P_1 spectrum weighted cross sections from NULIF along with the physical and geometrical description of the reactor core. Because of the large and diversified nature of the computations that are run on PDQ07, the discussions of each particular calculational method are explained in several topical reports. These include BAW-10116[14], on fuel assembly calculations and the use of fitted nuclear data, BAW-10117[2], a user's manual for B&W's version of PDQ07, and BAW-10118[15], on techniques and procedures for core calculations, BAW-10119[16], on power peaking uncertainty, and BAW-10120[17], which compares core physics calculations to measurements, BAW-10121[18], which explains the reactor protection system limits and setpoints, BAW-10122[19], normal operating controls, and BAW-10123[20], the incore instrumentation system. Since the details of specific calculations are discussed in these reports, no other additional explanations of the PDQ07 procedures and design capabilities are given here.

2.2.1.3 ANISN

Subsection 2.2.2.4 explains the ANISN[3] code.

2.2.1.4 DOT

DOT-IV[4] is a two-dimensional discrete ordinate code which computes the neutron flux in rectangular or cylindrical coordinates. Its other characteristics are the same as those in ANISN.

2.2.1.5 KENO

Subsection 2.2.2.1 explains the KENO[5] code.

2.2.2 Nuclear Criticality Codes

2.2.2.1 KENO4

KENO4[5] is a multigroup Monte Carlo criticality program written for the IBM 360 computer but converted to run on the CDC 7600. It is intended primarily to calculate the multiplication factor of systems of fissile material by providing a solution to the Boltzman neutron transport equation based upon the tracking of individual neutrons in the system. These neutrons are introduced by batches. The average of the batch K_{eff} values is considered to be the effective multiplication of the system.

The Monte Carlo technique allows development of a solution of the Boltzman equations for geometries which cannot be handled by other techniques. However, the Monte Carlo technique also introduces mathematical uncertainty into the solution which is not present in other techniques. KENO4 produces as part of its printed results the data necessary to evaluate this mathematical uncertainty. It shows the standard deviation of the average effective multiplication factor calculated. If one represents the KENO4 results by $\bar{K}$ and the real value of multiplicative factor by K_R then there is a 67 percent probability that $\bar{K}$ is within plus or minus one standard deviation of K_R and there is a 95 percent probability that $\bar{K}$ is within plus or minus approximately two standard deviations of K_R. This uncertainty is considered in all use of KENO4 results. This uncertainty is in addition to uncertainties concerned with cross-sections and physical modeling.

2.2.2.2 NITAWL

NITAWL[6] is an acronym for Nordheim's integral treatment and working library production. The function of this code is to read cross-sections written in the master library format, perform a resonance self shielding calculation when required, and to arrange the data in the format required by the code to be used in the neutronics calculation. NITAWL can produce cross sections for the ANISN, DOT, MORSE, XSDRNPM or KENO4 codes. NITAWL includes provisions for performing resonance calculations using the Nordheim's integral treatment - the most rigorous procedure. The narrow resonance and infinite mass approximations are available as alternate methods.

NITAWL is a unit of the AMPX[7] code package and processes cross sections in the AMPX format. This allows generalized treatment of the cross sections with the ability to develop specific reaction cross sections as desired. Temperature dependance of cross sections may also be included but in many cases the temperature dependent data available are limited in scope.

2.2.2.3 XSDRNPM

XSDRNPM[8] is a discrete ordinates code which can provide spectral and spatial averaging of cross sections. It provides a one-dimensional transport capability for calculating reaction rates, eigenvalues, critical dimensions, and allows spatial cross section weighting to be performed. The code may be used either to calculate the multiplication factor for simple geometries or to spatially weight cross sections to be used in the calculation of multiplication factors for more complex situations.

2.2.2.4 ANISN

ANISN[3] is more commonly used by B&W than XSDRNPM and is a one dimensional SN transport code which solves the Boltzman equation in plane, spherical, or cylindrical geometry. The name was derived from ANIsotropic SN because the incorporation of general anisotropic scattering in ANISN provided a major improvement over existing SN transport codes.

The SN (discrete ordinate) technique consists of dividing all possible directions of motions of neutrons at a given point into a manageable number of discrete units. The number of direction divisions considered can be varied so that the individual situations can be handled in the most simple manner possible. The solution to the Boltzman equation obtained in this way is energy, angle, and spatial dependent. As the SN order is increased (i.e., as direction of motion is divided into a large number of smaller angles) the solution approaches a rigorous description of the system involved.

2.3 Benchmarking of the Codes and Methods

2.3.1 Reactor Criticality Benchmarks

The benchmarking of the reactor criticality codes involves comparisons of the calculated results to the calculations or data from six other sources.

- Exact analytical solutions of simplified systems.
- Measured activation data, such as resonance captures and flux profiles.
- Experimental measurements of criticality and normalized power distributions.
- Reference calculations that are nearly exact for small systems.
- Tests and operational data from reactors.
- Tests conducted on fuel and material extracted from operating reactors.

The results of benchmarks are given in References 17 and 21. The criticality calculations show an overall standard deviation of $\pm 0.0017\ \Delta K_{eff}$ with measured data.

In Sections 3.0 and 4.0, it will be shown that the most reactive damaged fuel configurations are uniform arrays of stacked cylindrical pellets. These arrays are one of the simplest of all configurations to analyze. The critical buckling will be the same as the geometrical buckling allowing the geometrical buckling to be used directly with NULIF to determine criticality. When the geometrical configurations of the fuel arrays are simple shapes, such as cylinders or hemispheres the buckling can be analytically computed. For more complex shapes, spatial codes like PDQ or KENO are used to determine the bucklings. Since the arrays analysed for TMI-2 criticality calculation of the damaged fuel are uniform arrays of simple geometries, the following discussion gives a brief summary of the NULIF results for uniform critical experiments.

2.3.1.1 Uniformly Loaded Critical Experiments

Reference 21 describes 17 cold, clean uniform lattice critical experiments containing UO_2 fuel and H_2O moderator and 14 cold, clean uniform lattice critical experiments containing PuO_2-UO_2 fuel and H_2O moderator. All these configurations are approximated by right circular cylinders.

The NULIF-PDQ07 calculational results for each of these 31 critical assemblies are given in the reference.[21] The mean eigenvalue computed for the 17 UO_2 critical assemblies was 0.9983 with a standard deviation of 0.0047. For the 14 PuO_2-UO_2 critical assemblies, the computed mean eigenvalue was 1.0001 with a standard deviation of 0.0048.

2.3.1.2 Nonuniformly Loaded Critical Experiments

The undamaged portion of the TMI-2 core is like a standard core with control rods inserted. While the benchmark of the reactor codes is described in the safety analysis report[22], the results of nonuniform critical experiments with control rods is briefly reviewed below.

Reference 21 lists the core identification of 44 nonuniform, cold, clean critical experiments containing UO_2 fuel pins, water moderator, and various types and amounts of poison material. These lattices were designed to particularly mock up poison rods in the guide tubes of fuel assemblies of B&W reactor cores.

The NULIF-PDQ07 system was used to analyze these critical configurations many of which required three-dimensional representation. The results of the critical eigenvalue calculations are given in Reference 21. The mean eigenvalue computed for the 44 nonuniform loaded lattices was 0.9967 with a standard deviation of 0.0021.

2.3.2 Nuclear Criticality Benchmarks

2.3.2.1 ANISN

Several sets of benchmark calculations have been run using ANISN with the Hansen-Roach 16 group cross section set. Two of these are quoted as representative.

Effective multiplication factors for critical slabs of 3 weight percent enriched UO_2 and water are shown in Table 2-1. These critical thicknesses are taken from Reference 23. These results are quoted as a function of the order of angular quadrature. Note that for this case increased angular quadrature does not change the result.

Figure 2-1 shows the results of calculations of the infinite multiplication factor by ANISN.[24] The line labeled "PCTR DATA" is experimental values for the infinite multiplication factor for the mixture in question. The dashed line shows the infinite multiplication values calculated by ANISN. The results show good agreement.

2.3.2.2 NITAWL - KENO4

The NITAWL - KENO4 combination has been benchmarked by comparison with critical experiments in 21 cases representing the separation of fuel assemblies in water. In this work criticial experiments were performed with simulated fuel assemblies placed at various spacings. Details of this work may be found in Reference 25. The results of the comparison are shown in Tables 2-2 and 2-3. In cores I through X only moderator separated the simulated fuel assemblies. In cases XI through XXI sheets of stainless steel or borated aluminum were placed between the units to reduce interaction.

2.3.2.3 NITAWL - XSDRNPM - KENO4

In this combination of codes NITAWL is used to perform resonance self shielding calculations. XSDRNPM is used to spatielly weight cross sections, and KENO4 is used to calculate the system multiplication factor. This combination is used where the geometry is such that it would be difficult to describe each region of fissile material explicity. This combination has been benchmarked by applying it to a series of experimental measurements as described in Reference 26. This work was similiar to that described in Section 2.3.2.2.

However, the fuel bundles used contained fuel pins arranged on a closer spacing than was the case in Section 2.3.2.2. In this work the objective was to simulate the storage of exposed fuel pins which had been removed from fuel assemblies and repacked. The effective multiplication calculated for these critical systems are shown in Table 2-4.

Table 2-1. Multiplication factors for Critical Slabs of 3% (by Weight) Enriched UO_2 and Water Calculated by ANISN

Case	^{235}U density, g/ℓ	Thickness, in.	k_e S_4	k_e S_8	k_e S_{12}
1	51.90	7.17	1.004	1.004	1.004
2	71.85	6.64	0.998	0.998	0.998
3	86.40	6.55	0.992	0.992	0.992

Table 2-2. Benchmark Core Loadings Adjusted to 145 cm Moderator Height

Core*	Spacing between arrays, pin pitches	No. of B_4C pins	Moderator boron conc, ppm	Moderator temp, C	K_e at 145 cm moderator height: Measured K_e	1σ	Calcu by KENO4 K_e	1σ
I	—	0	0	21	1.0002	0.0005	0.998	0.006
II	0	0	1037	18.5	1.0001	0.0005	1.007	0.004
III	1	0	764	18	1.0000	0.0006	0.999	0.004
IV	1	84	0	17	0.9999	0.0006	1.004	0.007
V	2	64	0	17.5	1.0000	0.0007	1.005	0.005
VI	2	64	0	17.5	1.0097	0.0012	0.998	0.004
VII	3	34	0	17.5	0.9998	0.0009	0.994	0.005
VIII	3	34	0	17.5	1.0083	0.0012	1.003	0.005
IX	4	0	0	17.5	1.0030	0.0009	0.987	0.005

Table 2-3. Benchmark Core Loadings Adjusted to 150 cm Moderator Height

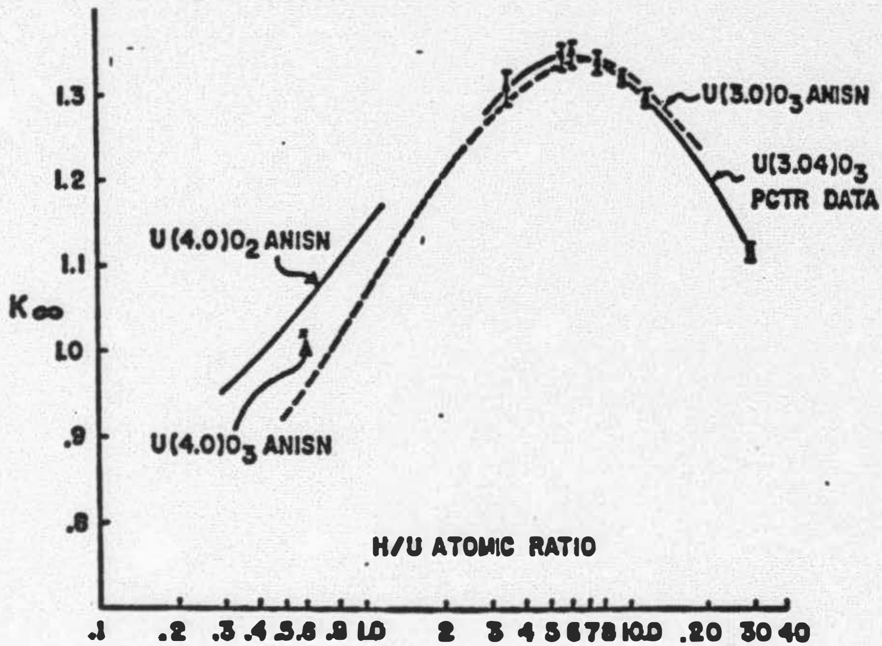
Core*	Spacing between arrays, pin pitches	Type of isolation sheets	Moderator boron conc, ppm	Moderator temp, C	K_e at 145 cm moderator height: Measured K_e	1σ	Calcu by KENO4 K_e	1σ
X	3	None	143	24.5	1.0001	0.0009	0.988	0.004
XI	1	S. steel	514	26	1.0000	0.0006	1.015	0.004
XII	2	S. steel	217	26	1.0000	0.0007	0.991	0.005
XIII	1	1.614% B/Al	15	20	1.0000	0.0010	1.008	0.005
XIV	1	1.257% B/Al	92	18	1.0001	0.0010	1.003	0.004
XV	1	0.401% B/Al	395	18	0.9998	0.0016	0.995	0.005
XVI	2	0.401% B/Al	121	17.5	1.0001	0.0019	0.990	0.005
XVII	1	0.242% B/Al	487	17.5	1.0000	0.0010	0.993	0.005
VIII	2	0.242% B/Al	197	18	1.0002	0.0011	1.005	0.005
XIX	1	0.100% B/Al	634	17.5	1.0002	0.0010	0.991	0.004
XX	2	0.100% B/Al	320	17.5	1.0003	0.0011	0.997	0.005
XXI	3	0.100% B/Al	72	16.5	0.9997	0.0015	0.981	0.004

*See Reference 25

Table 2-4. Calculated K_e for Arrays of Modules Containing Tightly Packed Fuel Pins

Core	Module arrangement	Intermodular spacing	Calculated K_e ± 1б
I	5 x 5	1.778 x 1.945	1.002 ± 0.007
II	5 x 5	2.538 x 2.709	0.984 ± 0.007
III	5 x 5	3.807 x 3.976	0.979 ± 0.006

Figure 2-1. Infinite Multiplication Factor Vs Moderation



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3.0 CRITICALITY CALCULATIONS PERFORMED FOR THE CORE REGION

3.1 Introduction and Background

The criticality calculations performed following the TMI-2 accident had the objective of determining the boron concentration which would ensure that the most reactive condition of the damaged core would be shutdown (where shutdown is defined by the standard technical specification[1] as $K_{eff} < .99$).

The NRC reviewed these calculations, along with their own and those of the Brookhaven National Laboratory, and concluded in their Safety Evaluation and Environmental Assessment[2] that the core will be subcritical in all possible configurations at about 3000 ppm boron. This conclusion was further supported by the NRC evaluation of core damage where they determined that only about 40% of the zircaloy cladding reacted with water.[3]

To ensure adequate shutdown margins in the core region during the proposed recovery operations (axial power shaping rod (APSR) insertion, control rod drive mechanism (CRDM) uncoupling, through head inspection of the reactor internals, and reactor vessel head removal) the criticality analyses performed immediately following the accident were reevaluated. The reevaluation of the criticality analyses considered the entire range of damaged core conditions that could credibly form a highly reactive configuration.

The discussions in Subsections 3.2 and 3.3 give the background of the analytical methods and procedures used for determining the most reactive condition of the damaged TMI-2 core. Subsection 3.2 provides an explanation of the parameters that affect the criticality analyses of the core. These parameters are grouped into four categories (i.e., fuel compositions). Discussions of the methods and procedures for analyzing each category are continued in Subsection 3.3.

Subsections 3.3 and 3.4 are included in this section as a recap of the methods and procedures used in the criticality analyses for TMI-2 during the cooldown period. They are primarily intended to show how those analyses were utilized in work reported in Subsection 3.5. It is not necessary to read Subsections 3.3 and 3.4 to understand the results of the current analyses.

The objective of the criticality calculations is to determine the most reactive condition (highest level of neutron multiplication) and show that it is subcritical by a safe margin. Thus, the methods and procedures used to determine the most reactive conditions involve many assumptions and approximations. These assumptions and approximations are intended to compensate for the uncertainties associated with the actual physical conditions (such as manufacturing tolerances or core damage in the case of TMI-2) by always being applied conservatively and hence producing a higher reactivity than would be realistic if they were not used.

The assumptions and approximations often tend to simplify the analyses. While the simplifications and conservatisms result in physical models that are not realistic, these models are not completely incredible. Thus, the modeling of the core (Subsection 3.4) centers on credible worst case assumptions and approximations. The methods and procedures for the analysis involve a systematic search through the categories of parameters for the worst case combination of parameters which produces the most reactive composition and configuration. The models of the damaged core that produced the most reactive conditions are discussed and schematically illustrated in Subsection 3.4. In Subsection 3.5 the results of the maximum credible damage configuration are presented.[4,5] This configuration is part of the design basis for the current safety evaluation[2] and for the reassessment of shutdown margin in this report. The results show that this configuration has a K_{eff} of 0.963. Subsection 3.5 also contains the results for the core K_{eff} (0.977) with all the fuel damaged. While this is a hypothetical configuration, it is worthwhile to consider when assessing the safety of the TMI-2 shutdown margins.

3.2 Parameters Affecting the Criticality Analyses

Criticality can be defined as a self sustaining neutron chain reaction. The objective of criticality safety is to determine the necessary conditions that will prevent a chain reaction from occuring, even when the system is in the most reactive condition possible.

The criticality analyses for TMI-2 began by identifying the parameters in the core that could affect criticality and produce a highly reactive configuration. Since neutrons will react with all isotopes, all the materials which comprise the core of the reactor affect the production and absorption of neutrons. Therefore to identify the materials in the reactor, the specifications for the TMI-2 core were reviewed.[6] The core region without fuel is an approximate cylinder 5.4 feet in radius and 13.8 feet high, filled with water. The water may contain other chemicals such as boric acid and lithium hydroxide. The fuel was in the form of 177 fuel assemblies, divided into 3 batches of 61, 60 and 56 assemblies. The assemblies within each batch were manufactured identically except for the difference in the U-235 enrichment. The assemblies are designed with a 15 by 15 array of cells on a square pitch. Each assembly contains 208 fuel rods with the 17 other cells consisting of one central instrument tube and 16 guide tubes. The guide tubes may contain either burnable poison rods, control rods, axial power shaping rods or water. The fuel rods contains stacked UO_2 pellets with zircaloy cladding and are surrounded by the water moderator. The cells within the assembly are held in a square array by Inconel grids[6].

In developing a list of parameters which affect criticality, one logical approach is to define categories of parameters which have similar effects. The following four categories are defined from the principal components of the neutron transport equation.

Fuel Composition
Moderator Composition
Fuel-Moderator Combinations
Fuel-Moderator and Relector Configurations

The production of neutrons originates in the fuel with the slowing down of the fission neutrons occurring in the moderator. The arrangement of the fuel and moderator affects the overall neutron absorption rate while the shape of the fuel-moderator configuration and the surrounding reflector affects the leakage rate.

Each of the following subsections discusses the effects that the various parameters have on criticality. The discussions are simply to demonstrate that a conscientious and systematic method was used to ensure that standard criticality practices are appplicable to the damaged TMI-2 reactor and that all parameters have been conservatively examined.

3.2.1 Fuel Composition

The parameters affecting criticality in the fuel composition category are:

- Chemical composition
- Fuel enrichment
- Burnup, or isotopic composition
- Physical reconfiguration with moderator and structural material
- Temperature

The fuel was manufactured in the form of UO_2 ceramic pellets with each pellet stack having the same weight and dimensions (within manufacturing tolerances). The isotopic content of the uranium was the only difference between pellets of the three fuel batches (1.98 wt.% U-235, 2.64 wt.% U-235 and 2.97 wt.% U-235).

The reactor operated at power for approximately 94 equivalent full power days. As a result of this operation, each fuel pellet achieved varying degrees of burnup in proportion to the average power it produced during that period. Consequently, each pellet contains plutonium and other transuranic isotopes as well as fission products. During the accident, the oxide fuel was exposed to an environment which could have altered its chemical composition. Therefore, the various fuel parameters considered as part of the criticality assessments included uranium, plutonium and transuranic content, fission product inventory, chemical composition, enrichment, and moderator and structural material that may have combined with the fuel or surrounded it.

The high temperatures during the accident could have resulted in oxidation of the UO_2, PuO_2, etc. to other oxide forms such as U_3O_8. However, only the dioxide form of the fuel needs to be considered in the criticality analyses since it has; (1) the lowest amount of oxygen, the lowest potential scattering, and lowest resonance integral;, and (2) the greatest density of uranium and is, therefore, the most reactive oxide form. While pure metallic uranium is more reactive than UO_2, the conditions necessary to convert UO_2 to U have not been shown to have occurred during the accident.[4,5] Consequently, the oxides, UO_2, PuO_2, etc. are the most reactive form to consider for criticality calculations.

Since some of the zircaloy cladding failed during the accident, the fuel was exposed to a high temperature steam and water environment. In this environment, the fuel solubility was considered along with chemical separation of the plutonium and fission products. However, uranium dioxide and the other transuranic oxides are neither soluble nor separable in the moderator environment which existed during the accident or has existed following the accident. To separate the constituents of the fuel in a reprocessing plant requires concentrated nitric and sulphuric acids used in a complex chemical process to dissolve the fuel with further chemical treatment to chemically separate the constituents.[7] The conditions necessary for this to have occurred at TMI-2 were not present. Thus, the chemical composition of the fuel is the dioxide form with uranium, fission products, and transuranics within ceramic particles.

The amount of uranium, transuranic elements and fission products in the fuel is a function of the initial enrichment and the burnup that each part of the fuel experienced. Before the accident, the three-dimensional core power distribution was continuously monitored and compared to predictions. The depletion of reactivity (indicated by the changing soluble boron concentration) was also being predicted. Therefore, the criticality calculation for the shutdown margin of the core has lumped the reactivity effects of burnup (uranium, plutonium, fission product content of the fuel, etc.) into one parameter.

The nominal fuel enrichment for each batch was used to determine the power and hence the burnup of the fuel along with the reactivity effects of burnup. As indicated above, this gives an accurate reactivity prediction. However, to ensure conservative criticality predictions, the reactivity of the fuel was calculated assuming that the manufactured concentrations of U-235 were all two standard deviations greater than the nominal values, 1.99 wt.%, 2.66 wt.% and 2.98 wt.%, respectively for batches 1, 2 and 3. So the reactivity effects of fuel enrichment were conservatively predicted to provide higher reactivities than would be realistic. Conversely, the total uranium mass was the nominal value for each batch. This combination of uranium mass and enrichments provides a conservative prediction of shutdown margin.

As a result of the accident, it could have been possible for the damaged fuel to be reconfigured into a fused mass with either structural material combined with the fuel or surrounding it, or moderator entrapped within the fuel. While these parameters will affect the reactivity of the fuel, calculations performed for this analysis show that moderating materials will increase the potential scattering within the resonances and thus decrease reactivity, and absorbing structural materials (even zircaloy) will decrease reactivity. Therefore, neither moderators nor structural materials has been considered in fuel particles that are less than the size of pellets.[7] Subsection 3.2.3, Fuel Moderator Combinations discusses the affects of zircaloy on amalgamated fuel particles that may be larger than a pellet. (No fuel particles of UO_2 larger than a pellet were considered since the temperatures necessary to melt UO_2 have not been shown to have occurred.[4,5])

The last parameter affecting the reactivity of the fuel is temperature. The temperature of the fuel is determined by the temperature of the surrounding moderator and the heat generation within the fuel. Because the resonance absorption of U-238 increases with increasing temperature, the lowest fuel temperature is the most reactive. Therefore, since heat generation will elevate the fuel temperature above the moderator temperature, it is conservative (more reactive) to consider the fuel temperature to be at the lower temperature of the moderator. Consequently, the fuel temperature will be the same as the moderator and the combined effect is discussed in Subsection 3.2.3.

In summary, the parameters in this category that will be directly treated in the criticality calculations are enrichment, burnup and fuel temperature.

3.2.2 Moderator Composition

Unmoderated uranium is not capable of achieving criticality at enrichments less than 5.0 wt% U-235.

Therefore, a moderator is necessary to produce criticality in an array of low enriched fuel similar to that in the TMI-2 core. Since the fuel will not dissolve in the moderator, homogeneous mixtures of the moderator and fuel are not considered. Furthermore, calculations performed for this study show that for low enriched fuel the moderator is much more effective surrounding the fuel. Therefore, the moderator will be considered as the medium that surrounds the fuel particles.

The as-built moderator region consisted of helium gas surrounding the fuel with zircaloy cladding encapsulating the fuel and helium to form fuel rods. The fuel rods are held in a 15 by 15 array with Inconel grids. Water, with other chemicals, surrounds the 208 fuel rods and fills the spaces not occupied by instrument tubes, guide tubes and other components,

The parameters that affect criticality in the moderator are:

- The density of the water (pressure and temperature).
- The boron concentration in the water.
- The other chemicals in the water used to control the pH of the water.
- Structural materials (Inconel, steel and zircaloy) and possible compounds of these materials.
- Fuel fines.

The density of the water is principally a function of its temperature and pressure. Following the accident the system pressure was approximately 1,000 psia and temperatures were 160°F or above. Once natural circulation was achieved, the temperatures allowed by the technical specification were between 280°F and 50°F. At temperatures below 160°F, less than a .3% error will result in the water density if the pressure is assumed to be atmospheric. Therefore, when temperatures are above 160°F, pressures of 1000 psia were used to compute the water density, and when temperatures are below 160°F, atmospheric pressure was used to compute water density. While the moderator contains boric acid, sodium hydroxide, lithium hydroxide and possibly, the hydroxides of the structural material (zirconium hydroxide, etc.), these compounds are in such dilute solution with the water that the density of water accurately reflects the density of the moderator.

The temperature of the moderator can affect criticality by, 1) changing the density of the water, 2) changing the velocity of the thermal neutrons, and 3) changing the atom density of the boron. Depending on the magnitude of these effects, the reactivity may either increase or decrease with an increase in moderator temperature. Since the moderator temperature effects will also be combined with the fuel temperature effects (both temperatures will be the same), the combined effects will be discussed in Subsection 3.2.3 where the fuel-moderator combinations are discussed.

The boron concentration in the moderator is defined as the weight percent boron in water. The control of the boron concentration provides the means of controlling criticality and ensuring an adequate shutdown margin.

Because the boron in the moderator results in a weakly acidic solution, strong hydroxides (lithium or sodium) in dilute concentrations are added to the moderator to control the pH factor. These hydroxides are all neutron absorbers and, therefore, decrease reactivity. However, since the concentration of the hydroxides is very dilute, their negative reactivity effect is ignored.

Neither the structural materials, nor oxides or hydroxides of the structural material will increase reactivity. These materials provide insufficient neutron moderation to offset the negative reactivity effects of their neutron absorption.[7] Calculations performed for this study have shown that only the zircaloy cladding has a low enough negative reactivity to be a factor in increasing the core reactivity by changing the relative amount of moderator in the mixture of fuel and moderator. This phenomenon was observed in the undamaged region of the core where the 50°F temperatures and 3500 ppm boron concentrations have produced a very positive moderator density reactivity coefficent. When the fuel in the undamaged core region was assumed to have no cladding and to be completely water logged, the reactivity decreased. This effect will be further explained when structural materials in the moderator are considered in the fuel-moderator combinations, subsection 3.2.3.

The one compound of the structural material that could affect reactivity is zirconium hydride. Because of the scattering properties of hydrogen and the very low absorption of neutrons by zirconium, this compound is a good moderator. It is present in very small quantities in any reactor with zircaloy clad fuel. During the accident, the hydriding of the zircaloy may have been accelerated by the high temperatures and free hydrogen that occurred as a result of the zircaloy reacting with water. However, assessments of fuel damage have concluded that the zircaloy was predominantly oxidized.[4,5] Therefore, the treatment of the zircaloy will be limited to a structural material of pure metal.

The last moderator parameter that could have an affect on criticality is the presence of fuel fines mixed in with the moderator. As a result of core damage during the accident, it is possible for fuel particles to be in the moderator region. They may be in various parts of the reactor coolant system including around areas of undamaged fuel. These particles may be in a range of sizes. While this effect may seem to conflict with the concept of having separate categories of fuel and moderator compositions, it will be explained further in subsection 3.2.3.

In summary, the parameters in this category that will be directly treated in the criticality calculations are moderator density, structural material, and boron concentration.

3.2.3 Fuel-Moderator Combinations

The discussions thus far have explained that low enriched fuel is not a criticality concern by itself. It is only through the effects of neutron moderation (slowing down the high velocity fission neutrons) that a sustained neutron chain reaction can be achieved. Therefore, the fuel and moderator must act in combination to increase the reactivity of the fuel. However, the combination that produces the highest reactivity will be very dependent on the boron concentration in the moderator.

This subsection considers the parameters that affect the criticality of the fuel-moderator combination. These parameters are:

- The size and shape of the fuel.
- The size and shape of the moderator surrounding the fuel.
- The structural material that may be in the moderator or surrounding the fuel.

When discussing the fuel and moderator compositions, it was explained that if a scattering material is in the fuel, it will increase resonance absorption and hence lower reactivity; it will also moderate the fast neutrons, increasing the probability of thermal neutron fission which will increase reactivity. These opposing effects can be eliminated if the moderator is placed around the fuel. Then the reactivity of the fuel is increased by neutrons being moderated without incurring the reactivity decrease caused by enhancing absorption in the fuel. Thus, lumping the fuel into particles enhances reactivity. Clearly, if the fuel were divided into particles so small that the fuel and moderator could be treated as if they were homogeneously mixed, this would not be effective. So a finite particle size that gives the largest reactivity must be determined. The limit of this particle size is a pellet, as long as the fuel contains no structural material which could cement pellets together. If structural materials are present in the fuel, as may be the case with the damaged TMI-2 fuel, then the maximum particle size is essentially unlimited. Therefore, fused fuel particles with a minimum of structural material needs to be assessed.

It will be explained in the Methods and Procedures Section (3.3) that the size of the particle which gives the highest reactivity when combined with the amount of moderator that gives the highest reactivity must be determined. Once this size has been determined, all fuel particles in the damaged part of the core will be assumed to have this size.

The shape of the fuel particles in the moderator also has an effect on reactivity. The pellets all started out as right circular cylinders. Power operation resulted in pellet cracking, and the effects of the accident probably fractured them with some fuel possibly becoming fused.

These irregularly shaped particles may be treated as regularly shaped particles with smooth surfaces if the volume of fuel is preserved and the ratio of surface area to volume, or the average chord length through the particle is preserved. While in reality it is not always possible to preserve the ratio of surface to volume, it is possible to create a parallelepiped, cylinder, or sphere that is equivalent to the irregular particle such that a regularly shaped particle results in a negligible reactivity difference. Thus, the particles have been treated as regularly shaped solids.

In addition to the particle's shape, the shape that a collection of particles forms in the moderator will also have an effect on criticality. If one section of the core is considered to be undamaged, small fuel particles may collect in the moderator around this undamaged section to affect overall reactivity. Futhermore, in the damaged section of the core it is possible for large and small particles to agglomerate together.

While an agglomeration of fuel with moderator filling any spaces between particles may be a realistic picture of the damaged core, the criticality modeling separates this agglomeration into regular and symmetric shapes. The moderator is modeled to surround the fuel with a volume that will be determined to provide the most reactive fuel-moderator combination. The fuel-moderator combination will have a rectilinear shape that is defined as a cell. In subsection 3.2.4, it is explained how the cells are formed into geometrical configuration by combining them in regular uniform arrays.

The undamaged portion of the core is considered to exist in cell arrays with each cell consisting of a cylindrical fuel particle surrounded a square moderator region. These cells are arranged in stacks that form fuel rods. The amount of moderator in this cell may not be the optimum amount to provide the highest reactivity. If this is the case, the structural material will either be the as-built amount or neglected, whichever is more conservative.

It is possible that fuel fines from the upper damaged region of the core are within the lower undamaged region (see Figure 1-1). Calculations performed for this study and by others have shown that if fuel particles of less than optimal size are added to particles of optimal size (with the fuel-moderator combination that produces the highest reactivity for each size particle) the resulting reactivity will be less than that of the fuel-moderator combination with optimally sized particles.[7,8] Thus fuel fines in the moderator are not explicitly considered in the calculations, rather all fuel will be assumed to have the same size as discussed in Subsection 3.2.2.

The temperatures of the fuel and moderator are considered to be the same. When the amount of moderator volume is being determined, the optimum temperature of the combination is a parameter that needs to be evaluated concurrently.

In summary, the parameters that affect criticality for combinations of fuel and moderator are, the size and shape of a regular fuel particle (parallelepiped, cylinder, or sphere), the amount of moderator surrounding the fuel in a regular rectilinear shape, the structural material in the moderator if the amount of moderator is constrained by the cell size (such as in the undamaged core), and the cell temperature.

3.2.4 Fuel Moderator and Reflector Configurations

The fuel-moderator combinations can be analyzed to determine the cell that is the most reactive. However, a single cell is so small (on the order of a few cubic centimeters) that the leakage of neutrons from this cell would preclude criticality. To attain the most reactive configuration, thousands of cells are assembled together. The parameters affecting the criticality of this configuration are:

- The size of the configuration
- The shape of the configuration
- The other material compositions that may be within the configuration
- The reflector material surrounding the configuration.

The number of cells to be considered in any configuration are the number possible from the total available mass of fuel. The shape of the configuration is one whose ratio of surface to volume is a minimum. The lower the surface area, the lower the leakage of neutrons and the higher the reactivity. The shape however, must be at least credible within the confines the TMI-2 reactor. For the portion of the core with undamaged fuel, the shape is the existing cylindrical shape; on the top of the undamaged portion may be a damaged portion also roughly resembling a cylinder.

The non-fuel materials that may be dispersed within the fuel cell array include the instrument tubes, the incore detector assemblies, the control rod guide tubes, (which may contain moderator, lumped burnable poison rods or control rods), and other structural material. The presence of these materials which absorb neutrons decreases the reactivity of regions where the fuel moderator combination has been optimized to achieve the highest reactivity. Therefore, in the damaged portion of the core, it is conservative to neglect these materials. In the undamaged portion of the core, the structural materials may displace moderator, thus the lumped burnable poison rods, instrument assemblies, etc. must be considered with respect to moderator displacement. However, their absorbing effects on criticality will be neglected. Only the absorption of the control rods is assumed to exist in the undamaged part of the core.

The reflector materials serve the purpose of returning neutrons leaking from the fuel-moderator configuration back to the fuel. By providing a fuel-moderator configuration with a reflector that completely surrounds it with optimal neutron reflection, the reactivity of the configuration will be increased. Many materials are good reflectors, so the criticality calculations must assess what materials in the core are the best reflectors, and whether they are distributed in a geometrically credible manner.

In summary, the parameters affecting criticality within the fuel-moderator and reflector configuration are the total mass of available fuel, the shape of the configuration, materials within the configuration that affect the moderator volume, control rods, and reflector materials.

3.3 Methods and Procedures

Section 3.2 provides an assessment of the parameters affecting criticality in each of the four categories that are identified as the principal components of the neutron balance equation for criticality:

- Fuel Composition
- Moderator Composition
- Fuel-Moderator Combinations
- Fuel-Moderator and Reflector Configurations

The basic parameters shown to be important in the criticality evaluation are:

- The enrichment of the fuel.
- Burnup of the fuel as a whole.
- Moderator density and the boron concentration.
- The size and shape of the fuel particles.
- The moderator volume surrounding the fuel.
- The structural material that could change the moderator volume.
- The temperature of the fuel-moderator mixture.
- The shape of the fuel-moderator configuration.
- Structural materials within the undamaged fuel-moderator configuration.
- The presence of control rods in the undamaged configuration.
- The reflector materials around the fuel moderator configuration.

The affects that these parameters, as well as others, have on criticality were assessed by considering each parameter separately. However, none of the parameters can actually be assessed independently when determining that parameter's affect on criticality. Each parameter has an affect on every other parameter. Thus, the methods and procedures used for the criticality calculations must provide a systematic optimization search of the parameters to produce the most reactive combined effect of all the parameters. Experience in criticality analyses has shown that conservative treatment of the parameters not listed above allows them to be independently assessed without affecting the optimization search.

This section describes the procedures for using the calculational methods discussed in Section 2.0 to determine the most reactive combination of all the parameters. These methods and procedures are the same ones used to ensure the TMI-2 reactor was safely shutdown (K_{eff} < .99) immediately following the accident. They have continued to be used in this study for the recovery operations.

The following discussions review the results of these systematic procedures showing how the most reactive combinations of the above parameters were produced to ensure that the reactor was subcritial. The results of this analysis are equally applicable to the safety evaluations for the proposed activities (APSR insertion, CRDM uncoupling, through head inspection and head removed) of the recovery operations.

The calculations of the TMI-2 core covered a broad spectrum of fuel configurations ranging from the intact core to homogeneous mixtures of moderator and fuel.

Calculations were performed on an undamaged core for which the enrichment of the fuel was the nominal manufactured value for the 3 batches of fuel. The burnup and reactivity of the fuel was precisely known as a result of calculations benchmarked to measured data. The moderator temperature was a variable ranging between 160°F and 280°F. The fuel was standard pellets within the standard 15 by 15 array. The cladding was assumed to be damaged and water could have filled the gap between the cladding and fuel pellets. The grids, however, were intact. The instrument tubes and assemblies were intact but damage to the lumped burnable poisons was assumed. The boron concentration was determined to produce a 1.0% $\Delta\rho$ shutdown margin.

Standard two-dimensional diffusion theory design and criticality calculations[6] were performed using NULIF-PDQ07[9] to determine the boron concentration required to keep the core 1% subcritical (K_{eff} = 0.99). The calculations showed that 1800 ppm boron was required, conservatively assuming that, (1) all control rods were out of the core, and (2) there were no poisoning effects from either the lumped burnable poison rods, xenon or samarium. (Lower moderator temperatures were found to be less reactive since the undamaged core has a positive temperature coefficient).

The positive moderator temperature coefficient in the undamaged core indicates that the damaged core could be much more reactive than the undamaged core. However, the arrangement of the fuel in the damaged core is unknown. Therefore, the calculation of the optimum combination of the parameters affecting criticality depended on ensuring that the core arrangement would be conservatively treated.

The parameters affecting criticality in the damaged core can be divided into 3 classifications; those that could be fixed, those that did not need to be a primary variable but whose affect could be applied as a reactivity coefficient, and those that were a primary variable.

The fixed parameters were enrichment and the reflector materials while temperature affects were analyzed at 160°F and 280°F. The parameters that could be calculated as coefficients are burnup, the structural materials in the moderator and undamaged core section, and the worth of the control rods. The primary variables included boron concentration, the size and shape of the fuel, the moderator volume, and the shape of the fuel-moderator configuration.

Before discussing the methods and procedures used to calculate the variable parameters, the procedures for determining conservative values of the fixed parameters will be explained as will the methods and procedures used to determine the reactivity coefficients.

The fuel in the undamaged core is arranged with the 1.98 wt. % batch 1 assemblies and the 2.64 wt. % batch 2 assemblies adjacent to each other in a checker board array in the central region. Surrounding the batch 1 and 2 assemblies on the periphery of the core are the 2.97 wt. % batch 3 assemblies. The neutronic importance of any batch is not significant since the assembly arrangement was selected to produce a relatively flat power distribution across the core. Therefore, a volumetric weighting of the batch, 1, 2 and 3 enrichments was judged to be a reliable means of averaging the enrichments. This average enrichment was increased by .05 wt. % to be conservative, resulting in 2.60 wt. % as the average enrichment of the fuel. KENO calculations with all fuel in the undamaged core having an enrichment of 2.60 wt.% were in good agreement with equivalent PDQ design calculations explicity representing each discrete assembly's enrichment.

In the damaged portion of the core, one average enrichment (2.60 wt. %) for all the fuel was also judged to be conservative since this would represent a random mix of the fuel. A scenario with the damaged batch 3 fuel collecting in the center of the batch 1 and 2 fuel and, thereby, increasing the effective average enrichment is considered incredible. Consequently, all damaged core models (including ones with a partially undamaged core) used a single conservative average enrichment of 2.60 wt. %.

The temperatures of the moderators under stable conditions were known from measured data. The highest value was 280°F while the lowest was 160°F. Therefore, both temperatures were analyzed at the most reactive respective conditions and a temperature coefficient was calculated to determine the reactivity effects of the temperature changes.

The effects of reflector materials are well known through experience. When critical arrays are surrounded by reflectors that are essentially infinite, the reflector savings of the materials is known to be the square root of their migration area. The reflector savings for any material or group of materials in the reactor is always less than 20 centimeters. Therefore, the leakage for any regularly shaped geometrical configuration is determined by the geometrical buckling with a 20 centimeter reflector savings. In cases where the reflector material was obviously known, such as the top and bottom of the core where the moderator is the reflector, and for the sides of the undamaged core where moderator and steel is the reflector, the migration length was used for the reflector savings.

The reactivity effects of the burned core had been accurately calculated by the design models as evidenced by the benchmarks to measured data. However, the KENO model with unburned fuel was used to determine the most reactive fuel configuration. Consequently, the effects of burnup on the reactivity were not part of the analyses to produce the most reactive condition. The reactivity effect of burnup was subtracted from the most reactive configuration once the results had been determined.

Calculations performed for this study show that the structural materials that would be in the moderator, such as the cladding, will decrease the reactivity of an optimum fuel-moderator mixture by .5% $\Delta\rho$. Therefore, structural materials in the damaged portion of the core were neglected. In the undamaged portion, the effect of structural material in the moderator and within the fuel-moderator cells was assessed with PDQ-NULIF calculations. The resulting effects on reactivity were applied to the most reactive KENO configurations.

The reactivity worth of the control rods had been calculated by the PDQ design model. Again, once the most reactive damaged core configuration had been determined with KENO, the worth of the control rods was subtracted from the criticality prediction.

The optimization analyses to determine the damaged fuel configuration that was most reactive, but contained enough boron to be 1.0% $\Delta\rho$ subcritical began by determining the most reactive moderator volume. The moderator density was determined at a temperature of 280°F and a boron concentration of 2100 ppm. The fuel was assumed to be stacked pellets. The fuel-moderator configuration was not specified, but the leakage was low enough not affect the neutron spectrum. The calculations employed both the reactor and nuclear criticality methods (NULIF, XSDRNPM, KENO-4).

While the moderator volume was the parameter of interest, the results were specified in terms of the fuel volume fraction. This is the ratio of the fuel volume to the total volume of the fuel-moderator cell. Thus, the larger the fuel volume fraction the smaller the moderator volume. The fuel volume fraction in a standard fuel assembly is .31. Since the undamaged core calculations showed that the fuel was overmoderated, the fuel volume fraction was increased. Initially, the fractions were somewhat arbitrary except that other criticality evaluations showed that the most probable fuel volume fraction for pellets randomly falling into a collection is .63; the volume fraction for pellets touching in a square array is .79 and in a hexagonal array is .91. The results of the analyses indicated an optimum value of .52.

Fuel-moderator configurations were analyzed with KENO. The configurations initially assumed all fuel to be damaged. The radial dimensions were assumed to be essentially infinite with fuel damage occurring axially, beginning at the top of the core. This modeling is discussed in detail in Section 3.4. The results indicated that leakage was minimal (less than 2.0% $\Delta\rho$) and that a total slump of all fuel into one mass would be most reactive. The most credible configuration within the core boundary is the right circular cylinder. However, the leakage of both the cylinder and a sphere were analyzed. The respective leakages, 1.6% $\Delta\rho$ and 1.2% $\Delta\rho$, were nearly the same.

Because of the low leakage of these optimal fuel-moderator configurations and the high reactivity of the fuel with a volume fraction of .52 at 2100 ppm boron, the boron concentrations were increased to 3000, 4000 and 5000 ppm. The moderator volume was again calculated to determine the most reactive combination with stacked fuel pellets. The results showed that, (1) with 3000 ppm boron a fuel volume fraction of .55 was most reactive, (2) with 4000 ppm boron a fuel volume fraction of .60 was most reactive, (3) with 5000 ppm boron a fuel volume fraction of .63 was most reactive. Figure 3-1 graphically illustrates these results.

The optimal moderator volumes at 3000, 4000, and 5000 ppm boron were then used to analyze the effects of pellet size on reactivity and to evaluate other fuel shapes that might be more reactive, such as sperical particles. The stacked pellets were shown to be the most reactive geometry.

The results show that 160°F is more reactive than 280°F, but otherwise the optimal parameters remained the same. Thus, the most reactive conditions and configurations for the damaged core region had been determined.

3.4 Damaged Core Models for Criticality Analysis

Core damage studies of the TMI-2 accident indicated that the upper region of the core was not covered by cooling water for an appreciable period of time.[4,5] During this time overheating could have resulted in substantial damage to the uncovered portion. This damage is projected to have been most extensive near the upper center of the core where the overheating was estimated to be the greatest.

In developing the core models for criticality analysis immediately following the accident, it was assumed that damage began at the top of the core and migrated downward. To simplify the model it was assumed that whatever damage occurred at a given height occurred over the entire horizontal plane of the core. In order to cover all of the credible possibilities, a number of damage situations were postulated. These situations consisted of assuming various amounts of the core had lost structural rigidity and the fuel had collapsed onto the spacer grids.

In all calculations only fresh UO_2 fuel and borated water were assumed as explained in Subsection 3.2. Neutron absorption by structural or poison material and radial leakage of neutrons were neglected in all cases. Conservative assumptions about the amount of fuel present were made such that the calculations contained more fuel than that actually manufactured.

The calculational models were as follows:

3.4.1 Fuel Collapsed on Grid Model

Calculations of the fuel collapse model are shown in Figure 3-2, Case I. This geometry was based on the hypothetical assumption that all of the fuel between the grids may have lost its structural integrity and fallen to the next lower grid.

The fuel volume fraction in these slabs was varied from the normal core situation, .31 to .79, and the boron concentration in the moderator was varied from 1500 to 2500 ppm. This represented the range of fuel damage from essentially undamaged fuel to complete collapse onto the grids.

3.4.2 Collapsed Grid Model

It was then assumed that the grids failed, and the fuel collapsed from the top grid to successively lower grids. The fuel first collapsed onto the top grid and then this grid collapsed to the next one, etc. The resulting geometries are shown in Figure 3-2, Cases II, III, and IV. In this series of calculations the temperature was assumed to be at 160°F and the pressure at 900 psia. The boron concentration in the moderator was varied from 2100 ppm to 4000 ppm, with a fuel volume fraction of 0.63.

3.4.3 Partially Collapsed Core Model

The geometry shown in Figure 3-3 represents a partially damaged core. In this case the first three grid levels are assumed to have collapsed with a fuel volume fraction of 0.55. The core below the fourth grid is assumed to be undamaged. The moderator boron concentration was 3000 ppm, and the temperature and pressure were 280°F and 1000 psia respectively.

Due to limitations in KENO, the calculations could not explicitly model both the upper and lower mixtures of fuel in a single calculation. Therefore, two KENO calculations were performed to bound the model configuration. The first case assumed a standard cell pitch of 1.44 cm, but used a larger pellet diameter in the damaged region to produce a fuel volume fraction of .55. The second case assumed the standard pellet diameter in the damaged region with a cell pitch of 1.12 cm and a correspondingly smaller pellet diameter in the lower region of intact fuel rods. These two cases bound the results for the partially damaged core. The first case gave a K_{eff} .004 higher than the second case and was, therefore, used in the safety evaluation.

3.5 Summary of Results

The damage model with the highest reactivity was the total core collapse model (Figure 3-2, Case IV). Figure 3-2 gives a schematic of the core modeling assumed for KENO-4 calculations. As the NRC has noted in Reference 1 of their Safety Evaluation and Environmental Assessment[2,3], at 3000 ppm the B&W calculations show that 100% fuel damage will not be critical when the fuel is in its most reactive condition and the effects of burnup and structural material are accounted for in the analyses. Reference 1 of the safety assessment further states that only "About 40% of the Zircaloy cladding reacted with water. This region of severe oxidation was localized above the 4 to 6 ft. elevation and may not have included peripheral bundles." A later NRC study bounding the estimates of damage concluded that for the maximum damage scenario, "Embrittlement of cladding by oxidation occurred to a depth of between 6 and 7 feet from the top of the core in the center assembly, down to about 5 feet to 6 feet in most of the assemblies, and did not occur on the lowest power corner assemblies on the periphery."[4]

Therefore, the maximum damaged core model is the upper three grids failing and 50% of the fuel collapsing on the fourth grid (see Figure 3-3). Before discussing the reanalysis of this maximum credible damaged core model the previous calculational results have been summarized. It was assumed that; (1) the fuel below the fourth grid was standing in its original configuration; (2) the array was infinite in the x,y directions; (3) the damaged fuel pellets in the upper region were packed in an optimum fuel/water ratio with a fuel volume fraction of 0.55, and; (4) the standing fuel pellets in the lower region had a fuel volume fraction of 0.31, which is typical of a normal core.

The results with unburned 2.6 wt% ^{235}U and the moderator at 280°F with 3000 ppm boron (no poisons or structural material) are tabulated below.

Calculational Results for
Maximum Credible Core Damage Model

Conditions: three grids fail, 50% cladding failure
.55 fuel volume, 280°F, 3000 ppm

Cell Pitch	KENO Slab K_{eff}	Core K_{eff}
1.46	1.000	.934
1.12	.996	--

The core K_{eff} includes the effects of radial leakage, fuel depletion, and the worth of the control rods in the undamaged portion of the core. This was the conservative estimate of core criticality when the reactor went into natural circulation.

These results clearly indicate that while the hypothetical model of 100% damaged fuel may be just subcritical, the maximum credible damaged core is safely shutdown (K_{eff} < .99) by a large margin.

Revised calculations of the maximum credible damaged model were performed to reassess the most reactive conditions for the safety evaluation of the proposed activities (APSR insertion, CRDM uncoupling, through head inspection and head removal). These calculations used the same methods and procedures as described in Section 3.3, but different values for the parameters as described below.

1) While a single uniform enrichment adequately represents the uniform flux on neutron importance weighting of the fuel batches, the radial leakage in the undamaged region of the core will be much greater than in the damaged region. The damaged region has a very low leakage determined analytically from a uniform cylindrical configuration while the undamaged region has a high leakage determined from PDQ calculations of the core.

2) The manufacturing records for the TMI-2 fuel were reviewed. These records showed the maximum core enrichment (the nominal value plus 2 standard deviations) was 2.555 wt%. Thus the core average enrichment was determined to be 2.555 wt% rather than 2.60 wt% assumed for the early core analyses. Based on this, all current and future criticality analyses utilizing an average enrichment will use 2.555 wt%.

3) The optimization of the temperature in the fuel-moderator cell showed 50°F (the lower allowable limit) as the most reactive condition. Thus, the fuel-moderator conditions were based on the moderator at 50°F.

4) The boron concentration is at least 3500 ppm (including uncertainties) with measured values not less than 3700. Therefore 3500 was used in the calculations.

5) With the changes in the enrichment, temperature and boron concentration, the fuel volume fraction producing the most reactive fuel-moderator combination increased from .55 to .63.

6) The reactivity coefficient of structural material in the moderator, and between the cells of fuel and moderator was determined with 50°F temperatures and 3500 ppm boron. As

before, the structural material in the moderator of the optional fuel-moderator mixture was a reactivity penalty and, therefore, neglected. In the undamaged core the effect was found to increase reactivity as explained in Section 3.3

7) The reactivity worth of the core average burnup was evaluated to determine the uncertainty in the decaying fission product inventory. This inventory has been changing ever since the fission process was terminated. Experimental data along with calculations have indicated the reactivity worth of the fission products is greater 3 years after shutdown than at the time of core shutdown.[10,11] However, the uncertainty in this worth is unknown. Therefore, to be conservative, the fission product poisoning effects at the time of shutdown are continued to be used in the calculations.

8) The worth of the control rods was reduced to reflect 50°F moderator temperatures.

9) The determination of the size and shape of the most reactive particle considered whole pellets as well as larger cylinders of fused particles. The fused particles had less reactivity than the whole pellets. Therefore, the optimization of the other variables considered only whole pellets.

Table 3-1 summarizes the reactivity effects of these parameters. The volume and flux, or neutron importance weighting of these coefficients was based on the KENO calculation of damaged and undamaged core regions, and the total core. With 50% of the core damaged and 50% undamaged, the KENO calculations indicated the reactivity effects of the damaged region were 4 times more important than those in the undamaged region. Thus the damaged core region had an 80% weighting factor for reactivity changes while the undamaged region had only a 20% weighting factor. The net effect of the core reactivity changes is to reduce the KENO K_{eff} by 6.0% $\Delta\rho$, resulting in a conservative estimated core K_{eff} of .943.

This reenalysis shows that the maximum credible damage scenario for the core is safely shutdown ($K_{eff} < .99$) by a 5.0% $\Delta\rho$ margin. Since the proposed recovery operations will not intentionally disturb the fuel, this scenario continues to be the design basis for criticality safety evaluations.

There is one other scenario that further demonstrates the degree of criticality safety for the proposed activities through head removal. This scenario treats the perturbation of fuel fines sifting from the damaged core region to the undamaged region as a hypothetical total core collapse (100% damaged fuel). An optimization assessment of this total core collapse model shows the highest K_{eff} is less than .977 as indicated in the following table.

Total Core Collapse (Hypothetical) Model

Conditions: 2.55 wt.%, .63 VF, 50°F, 3500 ppm; K_{∞} = 1.018

Reactivity Changes	% $\Delta\rho$
Leakage (Sphere)	-1.3
Burnup	-2.8

Core K_{eff} = .977

These results, the maximum credible damage model and the hypothetical total core collapse model, conservatively show the TMI-2 reactor will continue to be safely shutdown for the recovery operations through head lift. In Appendix A the criticality analyses performed by others have been reviewed for comparison. All results show the reactor is shutdown and B&W's optimization search has obtained the most reactive conditions.

TABLE 3.1

CORE CRITICALITY ASSESSMENT FOR MAXIMUM CREDIBLE DAMAGE

(KENO CALCULATION @ 280°F, 3000 ppm, VF[a] = .55, K_{eff} = 1.000)

Description	Core Changes	Reactivity Changes[b], % Δρ Damaged Region	Undamaged Region	Total Core[c]
1	Radial Leakage	− .6	−3.5	−1.2
2	Enrichment 2.6 wt% → 2.555 wt%	− .4	− .5	− .4
3	Temperature 280°F → 50°F	+1.5	− .7	+1.1
4	Boron 3000 → 3500 ppm	−2.5	−5.0	−3.0
5	Optimal Fuel VF .55 → .63	+ .2		+ .2
6	Water logged fuel & structural material	0.0	+6.1	+1.2
7	Burnup	−2.8	−1.3	−2.5
8	Control rods		−7.2	−1.4
9	Fused Fuel 3.4 Pellet Volumes with 10% zircaloy	0.0		0.0
	Total	−4.6	−12.1	−6.0

Estimated Core K_{eff} = .943

[a]VF, Volume fraction of fuel

[b]Reactivity changes are calculated with design models, NULIF-PDQ

[c]Combined 80% damaged with 20% undamaged (volume and neutron importance weighting; rounded to two digits)

FIGURE 3-1 REACTIVITY VS FUEL VOLUME FRACTION FOR FUEL PELLETS UNIFORMLY DISPERSED IN BORATED WATER

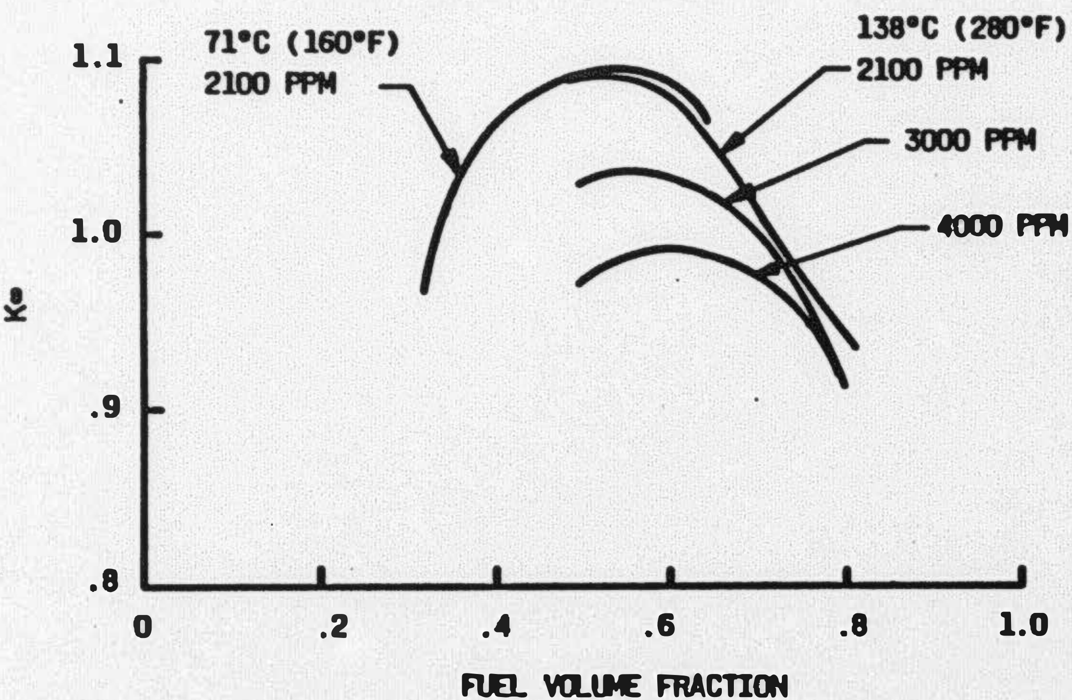


FIGURE 3-2 SCHEMATIC OF TOPPLED FUEL PELLET STACKS AND COLLAPSED GRID SUPPORTS

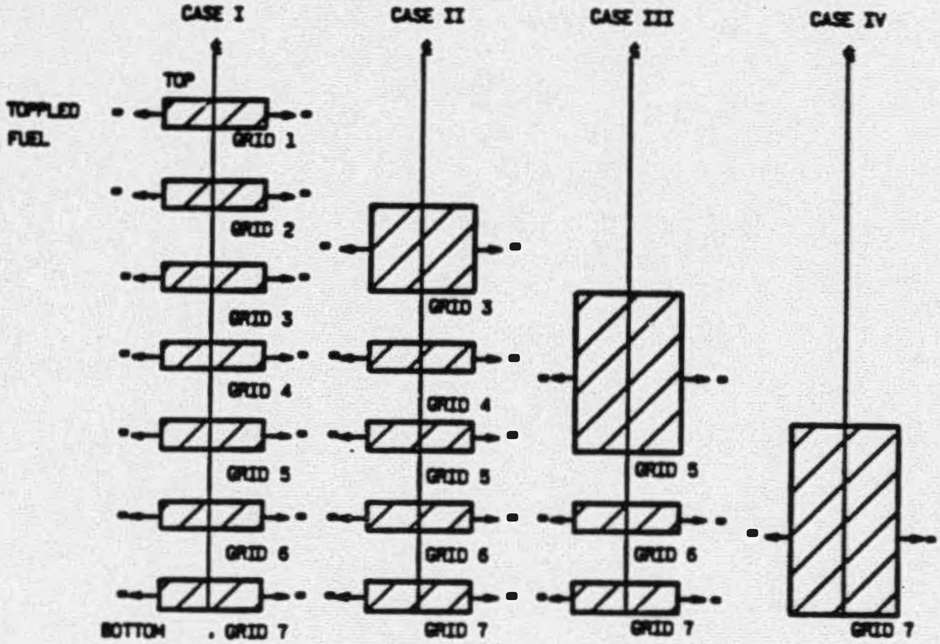
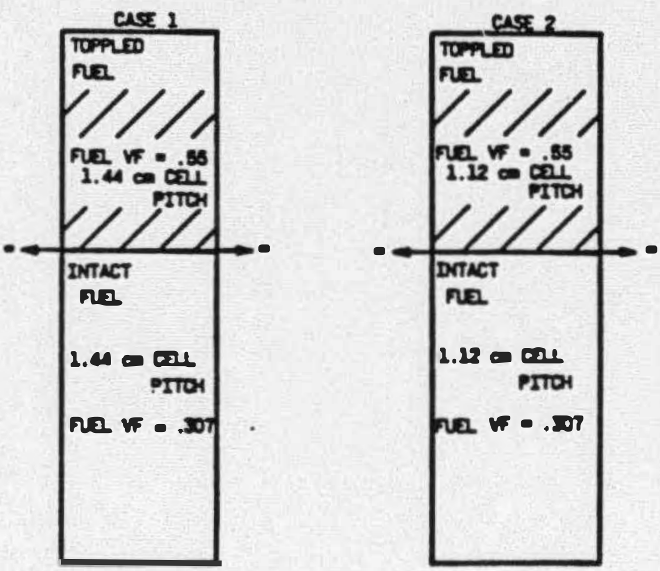


FIGURE 3-3 CONFIGURATION FOR THREE (3) GRID COLLAPSE



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4) Bounding Estimates of Damage to Zircaloy Fuel Rod Cladding in the TMI-2 Core at Three Hours After the Start of the Accident, March 28, 1979, M. L. Picklesimer, NRC Memorandum, June 20, 1979.

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4.0 ADDITIONAL CALCULATIONS FOR POTENTIAL FUEL ACCUMULATIONS OUTSIDE THE CORE

4.1 Introduction and Background

Section 3.0 reviews the criticality analyses performed for the core following the accident. These analyses, which form part of the bases for the current Technical Specifications[1] and Safety Evaluation and Environmental Assessment,[2,3] were updated to ensure the proposed recovery operations (APSR insertion, CRDM uncoupling, through head inspection and head removal) contained sufficient shudown margin. Not only was the maximum credible damage scenario reassessed, but the hypothetical perturbation of total core collapse was also reassessed. This section (4.0) is similar to Section 3.0, but the emphasis is to review and update the criticality analysis for fuel particles that are outside of the core region. The previous criticality calculations for fuel accumulations outside the core, showed the reactor vessel had greater potential to be a criticality concern than the fuel within the core. Therefore the objective of this section is to develop the maximum credible damage model for fuel outside the core and determine the conditions that ensure it will be safely shutdown for these recovery operations.

The following discussions provide an assessment of the criticality calculations performed for fuel particles in the lower vessel. This assessment includes both the particles that may have been transported throughout the system and collected in the vessel bottom once pump operation was terminated and the particles that may sift down to the vessel bottom during these recovery operations. The format for the discussion is the same that was used in Section 3.0. In Subsection 4.2 the parameters affecting criticality are reviewed based on the discussions in Subsection 3.2. The methods and procedures used for the analysis are explained in Subsection 4.3 and the results summarized in Subsection 4.4.

4.2 Parameters Affecting Criticality Analyses

In Section 3.2 four categories were defined which contain the parameters that affect criticality. These parameters fall into two broad groups, (1) those directly treated in the optimization analyses to determine the most reactive conditions, and (2) those not directly treated, but whose values were specified to produce the most reactive conditions. The following discussions review the parameters directly treated in the optimization analyses for the core and explain why some of these parameters do not need to be directly treated in the analyses for fuel out of the core region.

4.2.1 Fuel Composition

The parameters in Subsection 3.2.1 directly treated in the criticality calculations are:

- Enrichment,
- Burnup, and
- Temperature.

Burnup will not be directly treated for the fuel out of the core region because the burnup of this fuel is unknown. The burnup of the fuel varies from assembly to assembly and over the length of each assembly. Therefore, the burnup of the fuel out of the core region is dependent on which fuel was transported out of the core. Since it is difficult to estimate which fuel this is, burnup for all fuel will be neglected. This is a conservative assumption because any amount of burnup depletes ^{235}U faster than ^{239}Pu and ^{241}Pu are built up, and the fission products increase the absorption of neutrons. Thus, fresh UO_2 has a higher neutron production rate and lower absorption rate than burned fuel, which means that fresh fuel is more reactive than burned fuel.[4,5]

4.2.2 Moderator Composition

The parameters in Subsection 3.2.2 directly treated in the criticality calculations are:

Moderator density,
Boron concentration, and
Temperature.

These three parameters will also be directly included in the criticality calculations for the fuel out of the core region.

4.2.3 Fuel-Moderator Combinations

The parameters in Subsection 3.2.3 directly treated in the criticality calculations are:

The size and shape of fuel particles,
The amount of moderator surrounding the fuel,
The shape of the fuel-moderator cell,
Structural material, and
Temperature.

The structural material within an optimal fuel-moderator combination will always decrease reactivity. In fact, just 10% zircaloy in an optimal fuel cell will decrease the reactivity by 1.0% $\Delta\rho$. Therefore structural materials will not be considered in the fuel-moderator cells that have an optimal fuel to moderator ratio.

4.2.4 Fuel-Moderator and Reflector Configurations

The parameters in Subsection 3.2.4 directly treated in the criticality calculations are:

The total mass of available fuel,
The shape of the configuration,
Structural materials,
Control Rods, and
Reflector Materials.

For the same reason described above in Subsection 4.2.3, the structural material within the fuel-moderator configuration will be neglected.. Control rods outside the core region will also not be considered.

4.3 Methods and Procedures

Section 4.2 reviews the parameters affecting criticality that were discussed in detail in Section 3.2 and explains why a few of these parameters are not applicable to the region outside of the core. The parameters that are important for the optimization search in the criticality calculations outside the core are:

- The enrichment of the fuel
- The moderator density and boron concentration
- The size and shape of the fuel particles
- The amount of moderator surrounding the fuel
- The shape of the fuel-moderator cell
- The temperatures of the fuel-moderator and reflector
- The total mass of available fuel
- The shape of the fuel-moderator configuration
- The reflector materials.

Each of these parameters not only directly affects criticality but also affects the way every other parameter affects criticality. Thus, the procedures used in the calculations must provide a systematic optimization search to integrate the reactivity effects of all the parameters to produce the most reactive combined affect.

This section describes the procedures for using the calculational methods discussed in Section 2.0 to determine the most reactive combination of all the parameters affecting criticality. These methods and procedures are the same ones used to ensure that fuel locations throughout the reactor coolant system and reactor vessel were at least 1.0% $\Delta\rho$ subcritical (K_{eff} < .99) immediately following the accident. They have continued to be used in this study for the recovery operations (APSR insertion, CRDM uncoupling, through head inspection and head removal).

The optimization procedures can be best understood by explaining the reactivity trends that produced the most reactive conditions. Therefore, the analyses which produced the values of the various parameters will be discussed below. The methods for the calculation relied on the NULIF code and the criticality equation,

$$K_{eff} = K_{\infty} e^{-B^2M^2}$$

to determine the most reactive values for the various parameters. The buckling was either calculated analytically or determined from KENO calculations of the most reactive configuration.

The enrichment of the fuel out of the core region following the accident was considered to be the maximum average value, 2.60 wt %. While no fuel will be intentionally disturbed during the proposed recovery operations, it is hypothetically possible that some fuel would be preferentially disturbed. Therefore, not only was the average fuel enrichment reanalyzed, but also the enrichment of the highest enriched batch 3 fuel was analyzed. (For both values, two standard deviations on the uncertainty of the manufactured data were added to the as built enrichments).

The boron concentration determined to provide an adequate margin of subcriticality in the core region following the accident was 3000 ppm as discussed in Section 3.5. This same concentration was found to provide a 1.0% $\Delta\rho$ shutdown margin ($K_{eff} < .99$) for fuel out of the core region as will be explained in Section 4.4. However, the current boron concentration is greater than 3700 ppm. The minimum concentration, with an uncertainty of at least two standard deviations is assessed to be 3500 ppm. Therefore 3500 ppm boron was used in these analyses.

Various fuel particle shapes have been previously evaluated in Section 3.2.3. However, this previous evaluation had concentrated on a collection of fuel particles settled into a mass. In that case, the most reactive particle shape is a right circular cylinder with the cylinders stacked end on end such that the stack height is neutronically infinite.

The possibility of fuel particles falling from the core region during the proposed activities (surrounded by moderator rather than stacked) raises the question of whether the cylindrical particle is still the most reactive. Therefore, ANISN calculations were performed on particles surrounded by moderator. In this condition the spherical particle was found to be the most reactive. This is because spherical geometry minimizes the surface to volume ratio and therefore minimizes neutron leakage.

Figure 4-1 shows the affects of particle size on K_{eff} for the most reactive fuel-moderator conditions in the most reactive geometrical configurations. For either the cylinder or the sphere, a fuel particle with the radius of a standard pellet is the most reactive.

The optimum fuel-moderator mixture (with 3500 ppm boron) was found to be independent of fuel particle size, the fuel-moderator temperature or geometrical configuration. A fuel volume fraction of 0.55 or a hydrogen to uranium ratio of 2.36 gives the most reactive mixture of fuel and moderator.

The most reactive fuel-moderator and reflector temperatures are dependent on the fuel volume fraction, or hydrogen to uranium ratio, and the amount of boron in the moderator. However, the integration of these variables to produce the maximum reactivity showed the lowest allowable temperature, 50°F, is the most reactive.

Considering hemispherical configurations with a minimum leakage and optimal fuel-moderator combinations, calculations of the maximum credible damage model[6] showed that 100% of the damaged fuel (50% of the core) was less reactive than all the damaged batch 3 fuel. Thus, the design bases for the calculations of fuel out of the core was the maximum credible damage of batch 3.

Two geometrical configurations of batch 3 fuel were found to produce the maximum reactivity, an infinite cylinder of falling particles, and spherical segments of stacked particles. A schematic of these configurations is shown in Figure 4-2.

While there is no 4 by 4 array of batch 3 fuel assemblies in the core, a configuration of more than 16 batch 3 assemblies was assumed to form an infinite cylinder of falling spherical fuel particles. The optimum reflector for the cylinder was determined to be the moderator. This reflector was determined by analyzing various combinations of stainless steel and borated water to determine which one gave the largest reflector savings.

Once the particles had fallen, they were assumed to collect in spherical segments on the bottom of the vessel. The size of the spherical configuration was determined by assuming that it contains all the damaged[6] batch 3 fuel, with an optimum amount of moderator. On top of the segments the borated water was the best reflector; on the bottom, the pressure vessel was used as the support for the fuel. However, since the vessel has a thickness that is not effectively infinite to neutrons, the vessel was modeled as if it were placed in unborated water to produce the largest reflector savings.

4.4 Summary of Results

This section summarizes the results of four configurations for fuel outside the core region. The first is the most reactive configuration previously analyzed following the accident; it serves as part of the design basis for the current Technical Specifications.[1] The second configuration demonstrates the conservatism in the design basis for the criticality safety of the proposed recovery operations (APSR insertion, CRDM uncoupling, through head inspection and head removal) compared to the first configuration. The third configuration is the most reactive dynamic case which demonstrates that no falling fuel can increase reactivity beyond the shutdown criteria. And the fourth configuration has the highest reactivity attainable for the maximum credible damage model.[6,7] The fourth configuration consequently serves as part of the design basis for safety evaluations of the proposed recovery operations. Tables 4.1, 4.2 and 4.3 summarize the models, methodology and results that were obtained for these configurations.

4.4.1 Design Basis for Tech Specs[1]

Following the accident, fuel particles could have been lifted out of the top portion of the core and transported throughout the primary coolant system, including the reactor vessel, by the operation of the pumps. Consequently, the previous criticality calculations included analyses of all potentially highly reactive out of core regions.

The bottom of the reactor vessel was identified as the region that could accomodate the most reactive configuration because of its size. Since the previous calculations for the maximum credible core damage scenario (50% damaged and 50% undamaged),[6] had shown the core to be subcritical at 3000 ppm (Section 3.5), calculations to assess the shutdown margin of the fuel in the bottom of the reactor vessel were performed under similar conditions. At a temperature of 280°F, with a boron concentration of 3000 ppm, a fuel volume fraction of .55, and stacked pellets, the most reactive fuel had a K_∞ of 1.038 (for a complete description see Sections 3.2 and 4.2). The two configurations that have the lowest leakages are the sphere and (the more credible in the vessel bottom) hemisphere. The results of these calculations are as follows.

Criticality Assessment for Fuel Outside the Core

Conditions: 2.6 wt % U-235, .55 fuel volume fraction, 280°F, 3000 ppm, K_∞ = 1.038

	Damaged fuel (%)	Radius, cm	K_{eff}
Sphere	20	92	0.989
Hemisphere	40	123	0.989

Approximately 20% of the maximum credible amount of damaged fuel would be necessary to have K_{eff} of 0.989 in a spherical configuration and more than 40% of the damaged fuel is necessary to have a K_{eff} of 0.989 in a hemisphere in the vessel bottom.

As indicated by the NRC staff in Reference 1 of the Safety Evaluation and Environmental Assessment,[2,3] "Nearly all of the broken and oxidized fuel debris should remain trapped in the upper core region because the upper end fittings have a grillage that would act as a screen." Therefore, it is considered highly improbable to have more than 20% of the damaged fuel out of the core region and nearly incredible to have more than 40% out of the core region. Thus, 3000 ppm boron in the coolant was judged to provide sufficient margin for fuel located outside the core to meet shutdown criteria.

While these results clearly show that the reactor is safely shutdown with 3000 ppm boron, they also show that the lower vessel region is potentially more reactive than the core region where K_{eff} of 0.934 was calculated under similar conditions. Therefore, when assessing the possible perturbations that could result from the recovery operations (APSR insertion, CRDM uncoupling, through head inspection and head removal) it was considered conservative to assume that either the entire damaged core could collapse into the lower vessel or that the entire amount of damaged batch 3 fuel could be preferentially disturbed and collapse into the lower vessel region.

4.4.2 <u>100% Damaged Fuel In Vessel Bottom</u>

The optimization procedures discussed in Section 4.3 defined the most reactive conditions for the fuel-moderator combinations. These conditions were used when analyzing 100% of the maximum credible damaged fuel collapsed in the vessel bottom with the exception of particle size. Since the grids and fuel rod cladding are undamaged in the lower region of the core, fuel particles falling from the upper damaged region would have to fall between the grids and cladding.

Only particles less than one-quarter the size of a pellet (one-half the pellet radius) could fit through the largest opening between the undamaged grids and cladding. (Actually, a particle with a diameter no greater than .393 cm. will fit between the grid and fuel rod cladding, which is about four-tenths of the pellet diameter of .940 cm.) Consequently, the particle size was [illegible] quarter pellet. A hemisphere of this fuel (88.5 assemblies) wo[illegible] have a radius of 159 cm. With the reflector savings included for a[illegible] optimal reflector, the K_{eff} of this configuration is .976 as shown in Table 4-2.

4.4.3 Batch 3 Falling Out of the Core

The preferential disturbance of a batch 3 fuel assembly was considered to be possible as a result of the recovery operations. Therefore, the possibility of batch 3 fuel particles falling out of the core region was analyzed. The most reactive fuel, (discussed in Section 4.3) in a semi-infinite cylindrical configuration of 16 assemblies with the moderator serving as the reflector, (Figure 4-2) has a K_{eff} of .976.

4.4.4 Design Basis for Recovery Operations

The preferential disturbance of all batch 3 fuel assemblies such that the maximum credible amount of damaged batch 3 fuel[6] collects in the bottom of the reactor vessel is the most reactive configuration. Therefore, this configuration, which is symmetric spherical segments, is the design basis for demonstrating a safe shutdown margin for these recovery operations (APSR insertion, CRDM uncoupling, through head inspection and head removal). As shown in Table 4-2, the K_{eff} for this configuration, with optimized fuel-moderator and reflector conditions is .979.

TABLE 4-1

BASIS & DESCRIPTION OF MODELS ANALYZED
FOR OUT OF CORE REGION

MODEL	SIZE OF PELLET	GEOMETRY	ENRICHMENT	BASES
Design Basis for Tech Specs	Whole	Hemisphere	2.6	40% of Damaged Core (35.5 fuel assemblies)
Maximum Credible Design Basis for Recovery Operations*	1/4	Spherical Segment	2.98	100% Damaged Batch 3 fuel (19 assemblies)
Maximum Credible Fuel in motion	1/4	Semi-infinite Cylinder	2.98	4x4 array of falling Batch 3 fuel (16 assemblies)
Whole Core Perturbations	1/4	Hemisphere	2.56	100% of Damaged Core (88.5 fuel assemblies)

*APSR insertion, CRDM uncoupling, through head inspection, head removal.

TABLE 4-2

SUMMARY OF RESULTS FOR OUT OF CORE REGION

PARAMETER[a]	Semi-infinite Cylinder Batch 3 Falling	Spherical Segment Design Basis Recovery Operations	Hemisphere Design Basis Tech Specs	Hemisphere 100% Damaged Fuel
K_{eff}	.976	.979	.989	.976
K_{∞}	1.031	1.045	1.038	1.001
B^2	.001543	.001798	.001173	.0006935
M^2	36.28	36.28	41.34	36.52

(a)See Table 4-3 for Methodology Used to Calculate the Parameter

TABLE 4-3

METHODOLOGY USED TO CALCULATE PARAMETERS
FOR MODELS STUDIED FOR OUT OF CORE REGION

PARAMETER	Batch 3 Falling	Design Basis Recovery Operations	Design Basis Tech Specs	100% Damaged Fuel
K_∞	NULIF	NULIF	NULIF	NULIF
K_{eff}	ANALYTICAL	KENO	ANALYTICAL	ANALYTICAL
B^2	ANALYTICAL	KENO	ANALYTICAL	ANALYTICAL
M^2	NULIF	NULIF	NULIF	NULIF

Figure 4-1

K_{eff} As A Function Of Fuel Pellet Size

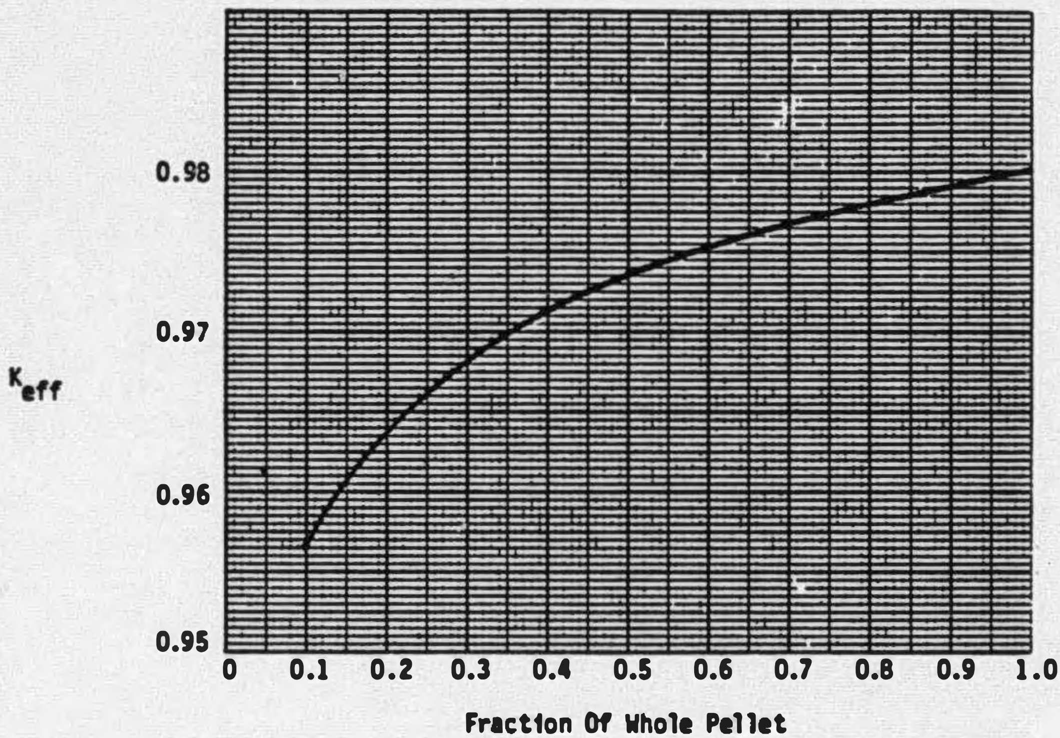


Figure 4-2

Schematic of Fuel Configurations

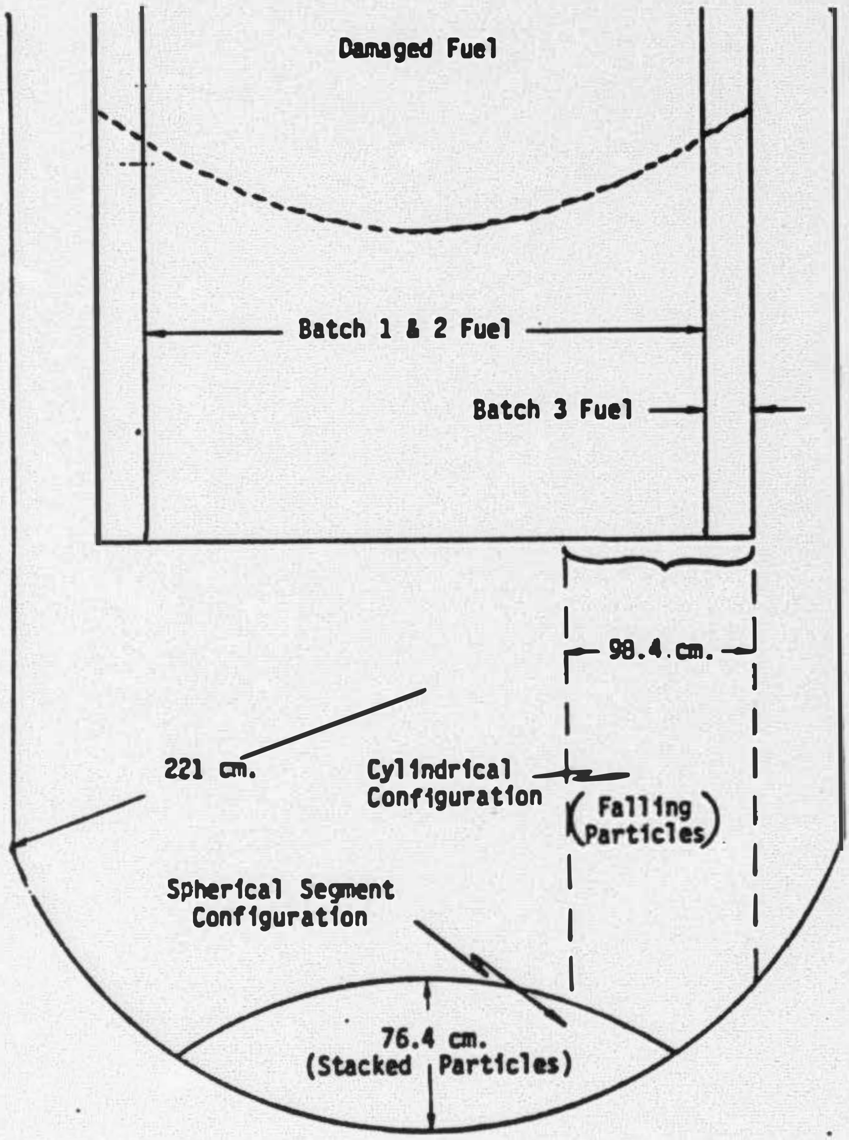
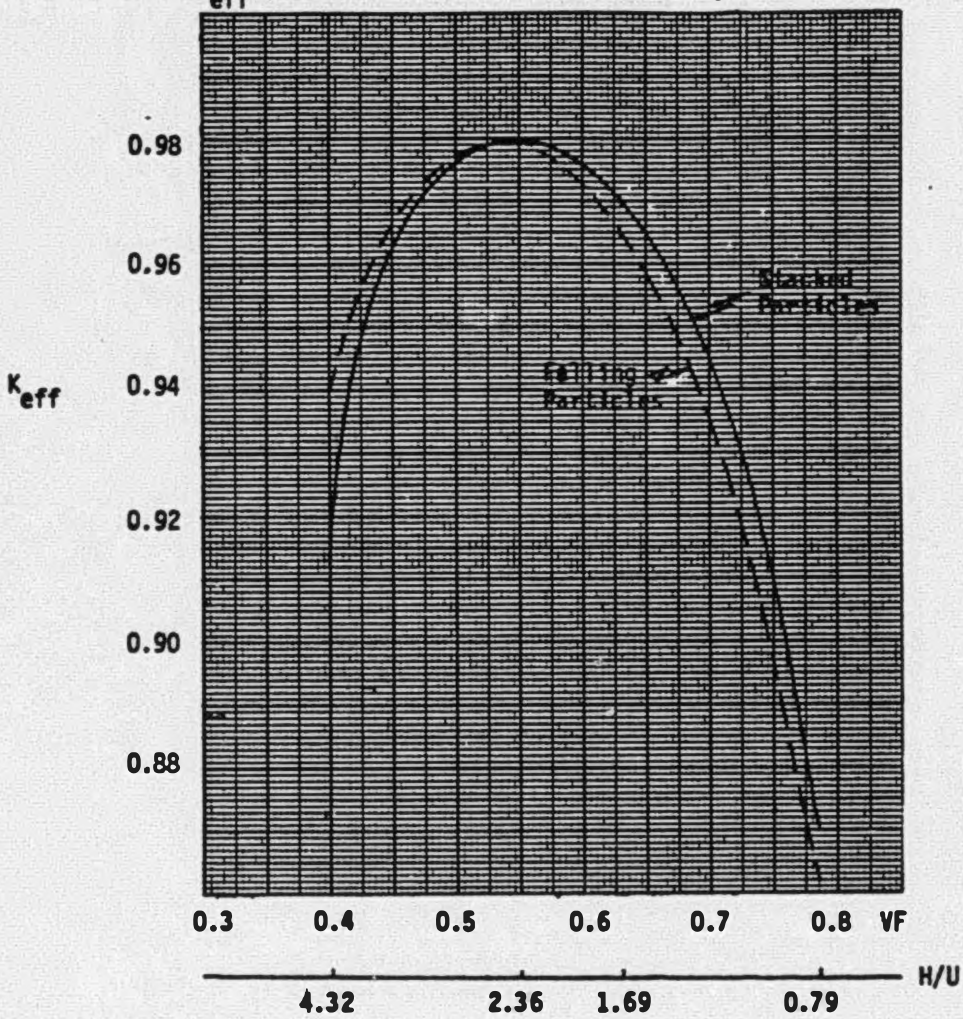


Figure 4-3

K_{eff} As A Function Of Volume Fraction (VF)

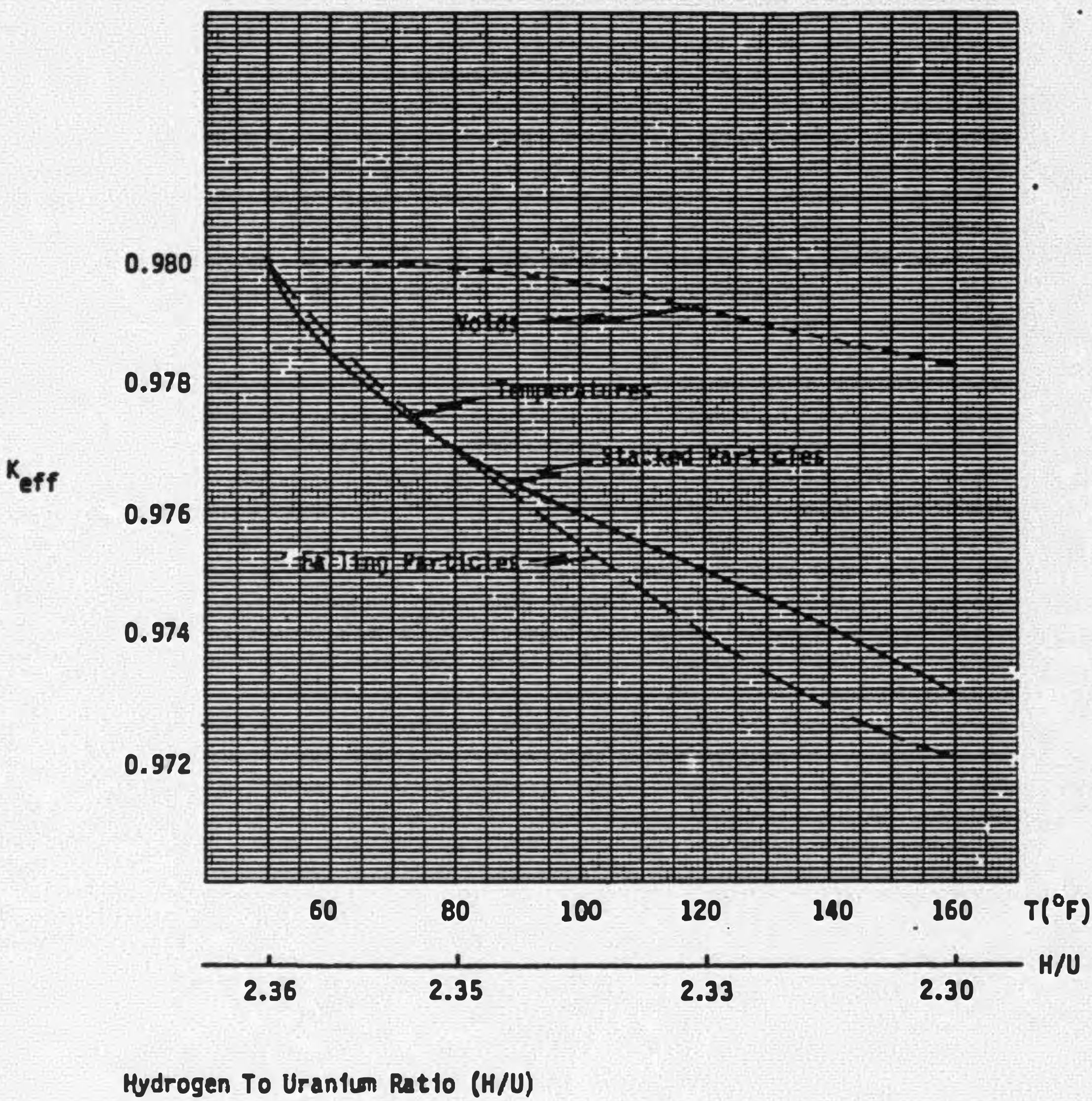


Volume Fraction (VF)

Hydrogen To Uranium Ratio H/U

Figure 4-4

K_{eff} As A Function Of Temperature And Moderator Void Density



Section 4.0 References

1) NUREG-0432, Three Mile Island Nuclear Station Unit 2, Technical Specifications, February 11, 1980.

2) NUREG-0647, Safety Evaluation and Environmental Assessment, Three Mile Island Nuclear Station, Unit No. 2, February 11, 1980.

3) NUREG-0557, Evaluation of Long Term Post Accident Core Cooling of Three Mile Island Unit 2, April, 1979.

4) NUREG-404, Generic Environmental Impact Statement on Handling and Storage of Spent Light Water Power Reactor Fuel, Volumes 1 and 2, U.S. NRC, March, 1978.

5) Docket 50-332, Revisions 01, 02 and 40, Barnwell Nuclear Fuel Plant - Safety Analysis Report.

6) Bounding Estimates of Damage to Zircaloy Fuel Rod Cladding in the TMI-2 Core at Three Hours After the Start of the Accident, March 28, 1979, M. L. Picklesimer, NRC Memorandum, June 20, 1979.

7) GEND-007, Three Mile Island Unit 2 Core Status Summary: A Basis for Tool Development for Reactor Disassembly and Defueling, D. W. Croucher, May, 1981.

5.0 SHUTDOWN MARGIN EVALUATIONS

In Sections 3.0 and 4.0 the maximum credible damage scenarios were evaluated for the core region and the out of core region respectively. For both regions it was shown that the shutdown criterion (K_{eff} < .99) is satisfied by a margin of at least 1.0$\%_{\Delta\rho}$ (K_{eff} < .98). In these sections, explanations were also provided describing how the most reactive conditions were obtained through an optimization of all parameters that affect criticality. However, there are two important additional considerations to be reviewed when assessing the shutdown margin of the reactor:

1) The degree of conservatism in the most reactive model compared to a more realistic model;

2) The uncertainty in the calculations.

The most reactive model was established by determining the highest reactivity for the maximum credible case. This model ensures a large margin of safety if it is much more reactive than a more realistic model and it meets the shutdown criteria. Therefore, in Subsections 5.1.2 and 5.2.2 the shutdown margins of more realistic models will be assessed for the core region and out of core region respectively to determine the margin of conservatism.

The uncertainty in the calculations is closely related to the degree of conservatism in the most reactive model. If the criticality predictions of the most reactive model do not significantly differ from the realistic model, then uncertainties determined from benchmarks of the calculational methods to measured data (that are similar to the realistic model) should be applied to the calculations. The uncertainty in the numerical and Monte Carlo solutions also should be applied to the calculational results. Subsections 5.1.3 and 5.2.3 discuss the calculational uncertainties for the core region and out of core region respectively.

5.1 Shutdown Margin in the Core Region

5.1.1 Maximum Calculated Criticality

Section 3.0 described how the maximum credible core damage model was analyzed to obtain the most reactive fuel conditions. In Subsection 3.5 it was shown that the shutdown margin of this model was 5.0% (K_{eff} = .943). In Subsection 3.5, the hypothetical perturbation of the total core collapsed into the most reactive condition and configuration was also discussed. The shutdown margin for total core collapse is 1.3% $\Delta\rho$ (K_{eff} = .977). Since the shutdown margin for the total core collapse model is much less than for the maximum credible damage model, the total core collapse model will serve as the basis for establishing the more probable core shutdown margin based on a more realistic assessment of fuel-moderator and reflector conditions and configurations.

5.1.2 Probable Shutdown Margin

The discussions in Subsections 3.2, 3.3 and 3.4 described how the values for the parameters affecting criticality were established to produce the most reactive combination of parameters and configurations. If the most reactive model is subcritcal by a significant margin, then the recovery operations (APSR insertion, CRDM uncoupling, through head inspection and head removal) can proceed with a confidence of safety. To illustrate the conservatism in the most reactive conditions, which is the hypothetical collapse of the total core, the reactivity affects of more realistic assumptions were considered. The following six items explain how the parameters listed in Table 5-1 were used to create a more realistic model from the total core damage model condition. Table 5-1 summarizes the reactivity effects and shows that the more probable value of K_{eff} for the core is less than .902.

1) Leakage

The configuration assumed in the model of total core collapse was a sphere surrounded by a water reflector. The more reasonable configuration for the total collapse is the cylindrical shape which formed the original core. In the spherical configuration it was also assumed that all fuel was uniformly mixed. If the damaged region collapsed into the undamaged region, the realistic distribution of fuel would be closer to the original fuel loading, rather than a uniform mix. Therefore, the cylindrical configuration was also considered to have a fuel distribution with the highest enriched fuel on the periphery. The total effect is to increase leakage and decrease reactivity by 1.8% $\Delta\rho$.

2) Distribution of Particle Sizes

It is not realistic to assume that the fuel particles in the core are solid pellets since during power operation the pellets crack. A more reasonable assumption is that the particles have a random distribution of sizes. This distribution will have an average reactivity affect that can be determined from Figure 4-1. The reactivity decrease will be more than .7% $\Delta\rho$.

3) Particle Arrangement

The collection of fuel particles cannot possibly be stacked end on end such as cylindrical pellets in fuel rods. A reasonable assumption is that the particles will collect in a random arrangement. Such a random distribution will allow ends of the cylindrical particles to be exposed to the moderator decreasing their reactivity by approximately 1.5% $\Delta\rho$.

4) Structural Material

The structural material was not included in the calculations to determine the most reactive conditions because it always resulted in reducing reactivity. Realistically, the zircaloy cladding, inconel grids, and other structural components will be in the core region in approximately the same proportion to the fuel as before the accident. Including the cladding and grids in with the fuel and moderator decreases reactivity by .8% $\Delta\rho$.

5) 50% Damage

The maximum credible damage model has 50% of the fuel damaged and 50% undamaged. In reality, the expected fuel damage is 40% clad failure. Assuming 50% damage to be credible rather than the hypothetical assumption of a total core collapse reduces the core reactivity by 2.5% $\Delta\rho$.

6) Actual Boron Concentration

Measured data indicates that at least 3700 ppm boron is in the reactor coolant system. Therefore, the assumed boron concentration of 3500 ppm used in the most reactive model could be realistically increased by 200 ppm. This additional boron is worth 1.2%$\Delta\rho$ in reactivity.

The total reactivity worth of these six changes to more probable conditions is a 8.5% $\Delta\rho$ decrease in reactivity. Consequently, while the hypothetical total core collapse model gives a K_{eff} of .977, and the maximum credible core damage model gives a K_{eff} of .943, the more realistic value of K_{eff} is less than .902.

5.1.3 Calculational Uncertainties

The calculational uncertainties to be applied to any criticality analyses are dependent on the following three factors,

- Criticality Criterion,
- Worst Case Model Assumptions, and
- Calculational Benchmarks.

Common practice is to neglect the application of uncertainties to the calculated results (except for the mathematical uncertainty in the computational model) if either, (1) the criticality criterion provides large margins of subcriticality (K_{eff} < .95) such that even reactivity accidents have essentially no probability of causing criticality, or (2) if the worst case model assumptions are so conservative that a large reactivity margin exists between the worst case calculations and reality. If the worst case model produces an effective multiplication factor that is within the calculational uncertainties of a realistic model and the criticality criterion is K_{eff} < .99, then an appropriate uncertainty factor, derived from calculational benchmarks to measured data, should be applied to the calculated results.

The calculational models used in analyzing the TMI-2 core involved both reactor and nuclear criticality methods as explained in Section 2.C. The reactor criticality and design methods established the most reactive fuel-moderator conditions in the damaged core region and the reactivity effects of burnup, control rods, etc. The nuclear criticality methods established the K_{eff} for the maximum credible core damage and total core damage configurations. Because the maximum credible core damage model was considered to be a conservative estimate of damage following the accident, no uncertainties were applied to the .934 calculated K_{eff} (Subsection 3.5).

More recent assessments of maximum and minimum core damage[1,2] do not show significant differences from the standpoint of criticality models. Therefore, it could be appropriate to consider an uncertainty on the KENO results for the maximum credible core damage model. The total uncertainty between KENO predictions of criticality and measured data using nuclear criticality methods is discussed in Subsection 2.3.2.2. This subsection refers to Tables 2-2 and 2-3 which give the results of twenty-one benchmark calculations. One standard deviation is approximately ±.01 ΔK_{eff} with the largest deviation in any case less than .02 ΔK_{eff}.

The KENO calculation of the core (Subsection 3.5, Table 3-1) gave a K_{eff} of 1.000 ± .005, where ±.005 is the standard deviation determined in KENO by the number of neutron histories. The total uncertainty in K_{eff} is the statistical combination of the mathematical and benchmark uncertainties,

$$\text{KENO} \quad K_{eff} = 1.000 \pm \sqrt{.01^2 + .005^2}$$

There is less than a 1.0% probability that the KENO K_{eff} is equal to or greater than,

$$\text{KENO} \quad K_{eff} = 1.000 + 2.33 \sqrt{.01^2 + .005^2}$$

$$\text{KENO} \quad K_{eff} = 1.026$$

where 2.33 is the 99% confidence limit for a one-sided uncertainty function. The core K_{eff} is 6.0% $\Delta\rho$ lower in reactivity than the KENO calculation as explained in Subsection 3.5. Therefore, the 1.026 can be used as a basis for determining a core K_{eff} that ensures, with 99% confidence, that the K_{eff} will not be greater than or equal to

$$\text{Core } K_{eff} = .967$$

5.2 Shutdown Margin for Fuel Accumulations Outside the Core Region

5.2.1 Maximum Calculated Criticality

Section 4.0 described the methods, procedures and calculational results for determining the optimum fuel-moderator and reflector combinations and configurations outside of the core region. The most reactive results were obtained for all of the damaged Batch 3 fuel (maximum credible case), stacked up to form rods in the bottom of the reactor vessel. The K_{eff} was .979. The following two Subsections, 5.2.2 and 5.2.3, respectively describe the shutdown margin for this worst case model when more realistic assumptions are modeled and when an uncertainty is applied to the calculational results.

5.2.2 Probable Shutdown Margin

The discussions of parameters affecting criticality (Subsection 4.3) explained how the most reactive combination of the various parameters was determined. If this most reactive combination is subcritical by a significant margin, then the safety of the reactor is assured during these recovery operations. To illustrate the conservatism of the most reactive conditions, the reactivity affects of more realistic assumptions were considered for the collection of Batch 3 fuel in the bottom of the reactor vessel. Table 5-2 summarizes the parameters and the reactivity worths. The following paragraphs explain how the parameters were changed from the worst case to the more realistic condition.

1) Distribution of Particle Sizes

Assuming that it is possible for only the damaged Batch 3 fuel particles to preferentially fall out of the core region, it is not realistic to assume that they will be all quarter pellets.

A more reasonable assumption is that the particles have a random distribution of sizes. The maximum diameter of fuel particles that could sift through the grids is .393 centimeters while the random distribution of particle diameters which could sift through the grid is 0.196 centimeters. This gives an average size of a cylindrical particle which is 1/23 that of the TMI-2 pellet. These smaller particles are more than 1.5% $\Delta\rho$ less reactive than the quarter pellet assumed for the worst case assessment.

2) Amount of Batch 3 Fuel Accumulation

In line with the above reasoning, it is not realistic that all the damaged batch 3 fuel could be transported to the bottom of the vessel. Reference 1 of the NRC Safety Evaulation[3,4] states that, "Nearly all of the broken and oxidized fuel debris should remain trapped in the upper core region because the upper end fittings have a grillage that would act as a screen." This also applies to the lower core region that is assumed to contain at least 3 grids intact holding undamaged fuel cladding that would act as a highly effective filter preventing fuel accumulation in the bottom of the vessel. Thus, rather than assuming all the damaged batch 3 fuel can collect in the bottom of the vessel, a more reasonable assumption is that 50% of the particles will reach the bottom. This will reduce the reactivity of the worst case by 3.2% $\Delta\rho$.

3) Particle Arrangement

Once the particles are in the bottom of the reactor vessel, there is no reasonable way for them to stack themselves end on end such as in fuel rods. A reasonable assumption is that the particles will fall into a random distribution. Such a random distribution will allow the ends of the cylindrical particles to be exposed to the moderator, decreasing their reactivity by approximately 1.5% $\Delta\rho$.

4) <u>Structural Materials</u>

If fuel particles are assumed to fall to the bottom of the reactor vessel, it is reasonable to assume that structural material will also fall. Adding zircaloy particles to the moderator in the same proportion to fuel and cladding that exists in the undamaged core will decrease reactivity by .5% $\Delta\rho$.

5) <u>Incore Instrument Guide Tubes</u>

The instrument guide tube structures extend down to the bottom of the reactor vessel. Since this structure will actually penetrate any collection of fuel in the vessel, it is reasonable to include its worth in a more realistic criticality analysis. The decrease in reactivity is at least .5% $\Delta\rho$.

6) <u>Batch 3 Burnup</u>

Even if the batch 3 fuel in the vessel bottom came from the least burned region of the core (i.e. the top two feet of the assemblies), it would have a lower reactivity due to the burnup. The minimum worth of the burnup is 1.0% $\Delta\rho$.

7) <u>Actual Boron Concentration</u>

Measured data indicates at least 3700 ppm boron in the reactor coolant system. Therefore, the 3500 ppm boron assumed for the most reactive conditions should realistically be increased by 200 ppm. This additional boron is worth 1.0% $\Delta\rho$ in reactivity.

The net effect of using more probable conditions for these seven parameters, rather than assuming worst case conditions, is a 9.2% $\Delta\rho$ decrease in reactivity. Consequently, while the calculated worst credible case has a k_{eff} of .98, the more realistic value of k_{eff} is less than .90 as shown in Table 5-2.

5.2.3 Calculational Uncertainties

The uncertainties in the calculational results arise from two sources. The first is the convergence criteria specified in iterative solutions such as PDQ or the number of neutrons specified in Monte Carlo solutions such as KENO. The second is the approximations inherent in the calculational models. The first uncertainty is easily determined since it is controlled by the user performing the calculations. The second uncertainty is more difficult to quantify because it embodies many different factors;

- The procedures used in the analysis,
- The uncertainties in benchmark measurement,
- The manufacturing tolerances on the fuel, and moderator structural components, etc.

As explained in Subsection 5.1.3, model uncertainties are not applied to the calculated results if the criticality criterion is large or if the worst case assumptions are very conservative. Furthermore, as explained below, the calculational models used in reactor criticality methods do not have a statistically significant uncertainty.

The uncertainty in the calculational models is inferred by comparing calculational predictions to measured data. Table 5-3, which is excerpted from the B&W Standard Safety Analyses Report[5], is a sample of the results from benchmark calculations used to determine the uncertainty in the basic NULIF modeling procedures. These results have a standard deviation of $\pm$ .0047Δ k_{eff}. This is not the uncertainty in the models, but the total uncertainty between the measured data and calculational models. In order to determine the uncertainty in the models only, the uncertainties in the measurements and the uncertainties resulting from fabrication tolerances must be statistically subtracted from the total uncertainty. The resulting modeling uncertainties have been found to be negative which means they are statistically insignificant. This has been confirmed with numerous benchmarks of both critical experiments[5] and operating reactor data[6] where measurement uncertainties have been shown to be greater than the deviations between calculated and measured results. Consequently, the modeling of the most reactive cases for TMI-2 simply requires an evaluation of the optimal configuration to ensure that it falls within the range of the calculational benchmarks. Since the most reactive configuration was a uniform array of fuel and moderator with a thermal spectrum similiar to a standard reactor, the calculational models are acceptable for the TMI-2 criticality analysis.

The KENO calculation of the most reactive configuration gave K_{eff} of .979 ± .00097. This small uncertainty can be increased by a factor of 2.33 and added to k_{eff}

$$k_{eff} = .979 + 2.33\ (.00097) = .981$$

to ensure, with 99% confidence, that k_{eff} will be less than .981 for the most reactive configuration that could credibly exist during the proposed first recovery operations.

5.3 Conclusion

The TMI-2 reactor has been safely shutdown since the accident. Any core perturbations that could result from the recovery operations (APSR insertion, CRDM uncoupling, through head inspection of the reactor internals, and reactor vessel head removal) will continue to be safely shutdown (K_{eff} < .99) by a 1.0% $\Delta\rho$ margin (K_{eff} = .98). The most reactive configurations within the reactor are as follows:

Case	Configuration	Fuel	k_{eff}
Max. Credible Out of Core	Spherical Segment In Bottom of Vessel	All Damaged Batch 3	.979
Total Core Collapse	Spherical Core Region	100% Core Damage	.977
Total Half Core Collection in Vessel Bottom	Hemisphere In Bottom of Vessel	100% Damaged Fuel	.976
Falling Fuel out of Core	Cylinder of Fuel Falling From Core To Vessel Bottom	16 Batch 3 Assemblies	.976
Maximum Credible In-Core	Cylindrical Core	50% Core Damage	.943

These configurations represent the optimum combinations of fuel-moderator and reflector conditions in a worst credible case scenario. If more realistic conditions are assumed, the above configurations will have shutdown margins of at least 10.0% $\Delta\rho$. Both the spherical segment of Batch 3 fuel in the bottom of the vessel and the spherical core with 100% damaged fuel were reassessed with more probable, and realistic conditions and both cases were shown to have k_{eff} less than .902.

Since the worst case scenarios contain large reactivity margins in comparison to more realistic conditions, the application of uncertainties to the calculated results is unnecessary. However, an evaluation of the uncertainties for the most reactive configuration of Batch 3 particles in a spherical segment in the vessel bottom shows that with a 99% degree of confidence of the K_{eff} will be less than .981. For the core region there is a 99% degree of confidence that the K_{eff} will be less than .967.

This evaluation of the worst credible scenarios for possible reactivity perturbations at TMI-2 during the above recovery activities demonstrates that there is sufficient shutdown margin to ensure criticality safety.

TABLE 5-1

REACTIVITY EFFECTS OF MORE PROBABLE CONDITIONS IN THE CORE

CONDITION	REACTIVITY % $\Delta\rho$
1) Leakage	-1.8
2) Random Size Particles	- .7
3) Particles In Random Arrangement	-1.5
4) Structural Material	- .8
5) 50% Fuel Damage	-2.5
6) 3700 ppm Boron	-1.2
Total	-8.5

Total core collapse K_{eff} = .977

Maximum credible damage K_{eff} = .943

More probable value of K_{eff} = .902

TABLE 5-2

REACTIVITY EFFECTS OF MORE PROBABLE CONDITIONS IN THE BOTTOM OF THE REACTOR VESSEL

	CONDITION		REACTIVITY % $\Delta\rho$
1)	Random Size Particles Falling Through Grids		-1.5
2)	50% of Falling Particles Can Go Through Grids		-3.2
3)	Particles Fall Into Random Distribution		-1.5
4)	Structural Material Falls with Fuel Debris		- .5
5)	Instrument Guide Tubes in Bottom of Vessel		- .5
6)	Burnup of Top 2 Feet of Batch 3 Fuel		-1.0
7)	Boron Concentration 3700 ppm		-1.0
		Total	-9.2

Maximum credible case value of K_{eff} = .979

More probable value of K_{eff} = .898

Table 5-3 Description of UO_2 Critical Experiments

Case[a]	U-235[b] wt %	Fuel density, g/cc	Pellet diameter, cm	Cladding thickness, cm	Lattice pitch, cm	H_2O volume fraction	Boron, ppm	Assembly radius,[c] cm	Buckling[d] 10^{-4} cm^{-2}	Deviation[f] ΔK
1	2.40	10.20	1.0298	0.0881 Al	1.6358	0.573	0	19.36	4.692	+.0049
2	2.40	10.20	1.0298	0.0881 Al	1.6358	0.573	864	32.91	3.702	+.0014
3	2.40	10.20	1.0298	0.0881 Al	1.6358	0.573	1536	68.77	4.601	+.0003
4	2.40	10.20	1.0298	0.0881 Al	1.5113	0.500	0	20.82	4.070	+.0078
5	2.40	10.20	1.0298	0.0881 Al	1.5113	0.500	1675	61.11	7.820	+.0010
6	2.70	10.18	0.7620	0.0688 SS	1.0287	0.452	0	32.01	5.400	-.0056
7	2.70	10.18	0.7620	0.0688 SS	1.1049	0.525	0	26.82	5.400	-.0012
8	2.70	10.18	0.7620	0.0688 SS	1.1938	0.593	0	24.27	5.400	+.0002
9	2.70	10.18	0.7620	0.0688 SS	1.4554	0.726	0	23.60	5.400	-.0034
10	4.20	9.46	1.1278	0.0400 SS	1.5113	0.498	0	18.75	3.599	+.0085
11	4.20	9.46	1.1278	0.0400 SS	1.5113	0.498	1152	26.09	3.063	-.0003
12	4.20	9.46	1.1278	0.0400 SS	1.5113	0.498	2342	38.36	5.251	+.0006
13	4.20	9.46	1.1278	0.0400 SS	1.5113	0.498	3389	59.71	3.920	+.0002
14	4.20	9.46	1.1278	0.0400 SS	1.4503	0.455	0	20.18	3.893	+.0049
15	1.30	10.53	0.9728	0.0871 Al	1.5570[e]	0.505	0	39.78	7.100	-.0062
16	1.30	10.53	0.9728	0.0871 Al	1.6510[e]	0.560	0	37.60	7.100	-.0065
17	1.30	10.53	0.9728	0.0871 Al	1.8059[e]	0.632	0	39.22	7.100	-.0071

(1) Reference 5
(b) UO_2 fuel enriched to specified wt % uranium-235.
(c) Plus a 30-cm H_2O reflector.
(d) Measured axial buckling.
(e) Hexagonal lattice; all others are square.
(f) Deviation in K_{eff} from the average of .9983;
+ is higher than average and - is lower, σ = .0047.

SECTION 5.0 REFERENCES

1) Bounding Estimates of Damage to Zircaloy Fuel Rod Cladding in the TMI-2 Core at Three Hours After the Start of the Accident, March 28, 1979, M. L. Picklesimer, NRC Memorandum, June 20, 1979.

2) GEND-007, Three Mile Island Unit-2 Core Status Summary: A Basis for Tool Development for Reactor Disassembly and Defueling, D. W. Croucher, May, 1981.

3) NUREG-0647, Safety Evaluation and Environmental Assessment, Three Mile Island Nuclear Station, Unit No. 2, February 11, 1980.

4) NUREG-0557, Evaluation of Long Term Post Accident Core Cooling of Three Mile Island Unit 2, April 1979.

5) B-SAR-205, Babcock & Wilcox Standard Safety Analysis Report, April 10, 1979.

6) BAW-10120, Comparison of Core Physics Calculations With Measurements, J. J. Woods, March, 1978.

APPENDIX A

REVIEW OF ANALYSIS PERFORMED DURING THE COOLDOWN PERIOD

Reactivity calculations performed by five separate organizations to evaluate nuclear criticality safety of the damaged TMI-2 core are summarisied and reviewed below. All these calculations were performed during the cooldown period and all were reported by December, 1979. Evaluations performed by the Babcock & Wilcox Company (B&W), the GPU Service Corporation (GPU), the Brookhaven National Laboratory (BNL), the Oak Ridge National Laboratory (ORNL), and the U. S. Nuclear Regulatory Commission (NRC) are reveived. Four of these organizations reported their work at the 1980 Winter Meeting of the American Nuclear Society. The evaluations were not totally independent from each other in that Oak Ridge and GPU relied on starting information supplied by B&W. One of the computer codes used by GPU was also used by B&W (PDQ07). One of the codes used by B&W, the NRC, and ORNL is the ORNL code KENO (although different versions were used). However, the evaluation strategies and accident models assumed were arrived at completely independent from each other.

The objective of each evaluation was either to determine the degree of subcriticality of the damaged TMI-2 core or to provide a basis for recommending the concentration of soluble boron necessary to ensure an adequate margin of shutdown reactivity for the damaged system. It must also be recognized that these calculations were made without definition of the exact system to be calculated; therefore, highly conservative assumptions were made and hypothetical configurations were evalusted to provide assurance of safety. These evaluations were intended to address the maximum credible reactivity and the safety of the damaged TMI-2 plant, and although not originally intended to demonstrate the safety of disturbing the damaged system, they do provide a basis upon which such evaluations can be carried out. B&W, GPU and the NRC reported that soluble boron concentrations of 3,500 to 4,500 ppm would be required to maintain at least a 1% shutdown margin assuming a hypothetical extreme condition of total core disruption at optimum reactivity conditions and assuming fresh fuel. BNL felt 3,000 ppm boron would be adequate. B&W (assuming a specific scenario of damage) reported that the damaged core could have K_{eff} as high as 0.932 while none of the postulated damaged models assumed by ORNL had a K_{eff} greater than 0.90. Direct comparisons between the different evaluations however are difficult, and where a direct comparison is possible there are the expected variations.

The calculations, taken as a whole, demonstrate that the damaged TMI-2 core was shutdown with K_{eff} probably less than 0.95. Taken as a whole, they also infer quite reactive hypothetical cases possibly requiring 4,500 ppm boron. However, in retrospect these hypothetical cases are not considered credible in light of new assessments of the core damage given in GEND-007, May, 1981.

Babcock & Wilcox Company (B&W)

B&W reported its work December, 1979 in NPGD-TM-434.[1] This work was also reported at the 1980 Winter Meeting of the American Nuclear Society.[2] Since B&W's calculations are reported in detail elsewhere in this report, only results will be summarized. After postulating a conservative configuration of damage and using the results of the computer codes NULIF,[3] PDQ07[4] and KENO[5] (with additional adjustments for fuel burnup), B&W concluded that K_{eff} of the damaged TMI-2 core could be as high as 0.932 at 3500 ppm boron (this evaluation is updatd in Section 3.5). B&W also concluded that 3,000 ppm boron could maintain this configuration at K_{eff} < 0.99. B&W, however, did identify hypothetical scenarios of damage that might require soluble boron concentrations of up to 4,500 ppm boron to maintain subcriticality. Those are now not considered to be credible in view of new assessments of core damage shown in GEND-007.

GPU Service Corporation (GPU)

GPU reported their work August 31, 1979 in the GPU report TDR-049.[6] This work was also reported at the 1980 Winter Meeting of the American Nuclear Society.[7] GPU used the reactor design computer code XPOSE[8] to perform parametric studies and to calculate K_∞. They used PDQ07 to calculate K_{eff}. Using XPOSE and PDQ07, GPU evaluated the reactivity of an undamaged core as a function of the soluble boron concentration and concluded that the undamaged core without any poison material except soluble boron had a K_{eff} of 0.940 at 3,400 ppm boron. GPU noted that its calculation of a specific reactor condition also calculated by B&W was 0.02 ΔK_{eff} lower than that calculated by B&W. Consequently GPU adjusted their PDQ results to B&W results to incure conservatism.

GPU calculated the multiplication factor for infinite media of both homogeneous and heterogeneous UO_2/H_2O fuel mixtures as a function of UO_2 volume fraction and soluble boron concentration. XPOSE was used to determine K_∞ assuming the middle core enrichment of 2.64 wt % without credit for any fixed neutron poisons; both UO_2 pellets and pellet fragments were considered. GPU's calculations of K_∞ by XPOSE[8] were compared to B&W's calculation of K_∞ using NULIF.[3] Although the agreement is excellent, a variation of 0.01 ΔK_∞ can be observed (GPU being higher at optimum UO_2 volume fractions). GPU reported several hypothetical configurations that were critical at a boron concentration of 4,000 ppm; however, if reasonable but conservative assumptions are made a boron concentration of 3,500 ppm would be necessary to maintain at least a 1% shutdown condition. Maintenance of a 3500 ppm boron concentration was recommended by GPU.

Brookhaven National Laboratory (BNL)

BNL reported its work by memorandum on May 18, 1979.[9] BNL used the HAMMER computer code[10] with input bucklings for leakage correction to calculate K_{eff} of a number of spherical systems. BNL assumed these systems to contain either 30, 50 or 100 percent of the TMI-2 UO_2 fuel inventory at the maximum average core enrichment of 2.6 wt% U-235. No credit was taken for any poisoning material except soluble boron. BNL used the results of a parametric study for each system versus volume ratios of UO_2 to H_2O and versus different concentrations of soluble boron to determine the concentration necessary to maintain the three fuel inventories subcritical.

Results are summarized below:

Percent Fuel inventory	Minimum boron concentration to insure subcriticality
30	2,720 ppm
50	2,900 ppm
100	3,060 ppm

The BNL memo acknowledges uncertainties associated with their analysis. BNL reports no benchmark effort to validate this methodology. The reported boron concentrations necessary for subcriticality are considerably lower than those reported in other work summarized in this document.

Oak Ridge National Laboratory (ORNL)

ORNL reported its work December, 1979 in ORNL/CDS/TM-106.[11] ORNL personnel gave two papers summarizing their work at the 1980 winter Meeting of the American Nuclear Society.[12,13] ORNL used the Monte Carlo computer codes MORSE-SGC/S[14] and KENO-IV[15] to model a number of possible damaged core configurations. ORNL used both codes to calculate K_{eff} for the hot, zero-power startup configuration of TMI-2 (actual $K_{eff} = 1.00$). K_{eff} by MORSE was 0.987 ± .003 and by KENO the calculated K_{eff} was 0.983 ± .006.

ORNL did not evaluate a TMI-2 core model assuming total core disruption. ORNL did evaluate three disrupted core models that simulated three damaged core configurations in an internally consistent manner. The most elaborate of these models which ORNL has designated as the "Three Jump Slump" model divides the TMI-2 core into eight different axial regions and four different radial regions. Several disruptive mechanisms such as fuel swelling, ZrO_2 in coolant, fuel slumping, etc. have been included in this model. ORNL did not search for a minimum critical soluble boron concentration. All evaluations were made assuming 3,180 ppm boron. For the intact portion of the reactor core, each of the three models includes a representation of the contents of the 39,825 pin lattice locations. The major difference between the three models was the number of axial layers used to represent the disrupted portion of the core. In each of the models, ORNL assumed that control rods and LPB rods remain in the intact portion of the core. The worth of these rods was determined for each model. Detailed descriptions of these models are available in the ORNL report, ORNL CSD/TM-106.[11]

Result of the ORNL study are summarized below.

Model	Base configuration	Base, no control rods
"Three Jump Slump"	0.862 ± 0.006	0.875 ± 0.006
"Displace-Fuel Slump"	0.845 ± 0.006	0.875 ± 0.006
"In-Place Fuel Slump" - maximum reactivity at 50% swelling	0.845 ± 0.005	

U.S. Nucler Regulatory Commission (NRC)

The NRC reported its work by memorandum on May 14, 1979.[16] The NRC work was also reported at the 1980 Winter meeting of the American Nuclear Society.[17] The NRC used the Monte Carlo computer code KENO[15] together with the 123-group GAM THERMOS library. The NRC modeled a number of configurations including an undamaged core; the most reactive, intact damaged latticed core; and a fully pelletized damaged core. The NRC calculated a known hot, zero power condition for TMI-2 to establish confidence in this modeling. The NRC reported that it calculated this condition to within 0.005 ± 0.004 ΔK_{eff} on the high side. The NRC performed parametric cell studies assuming both a 2.31 wt % and a 2.96 wt % U-235 enrichment. Core calculations assumed three radial zones with different enrichments. The NRC concluded that the core, intact, but at the most reactive pitch, will be subcritical at 3,000 ppm soluble boron concentration.

The NRC also evaluated K_{eff} for a spherical system assuming all fuel rods had ruptured, emptying all their pellets into a pile of hard UO_2 spheres with borated water in between. Fresh fuel at an enrichment of 2.57 wt % U-235 (core average) was assumed. The NRC reported a K_{eff} of 0.997 ± 0.004 at 3,500 ppm for this situation.

APPENDIX A

References

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11 R. M. Westfall, et al., Criticality Analyses of Disrupted Core Models of Three Mile Island, Unit 2, ORNL/CSD/TM-106, December, 1979.

12 J. T. West, et al., "TMI-2 Criticality Analysis - Analytical Models and Methods, Transactions American Nulcear Society, 35, 274-276, November, 1980.

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15 L. M. Petrie and N. F. Cross, KENO-IV - An Improved Monte Crlo Criticality Program, ORNL-4398, 1975.

16 Memorandum, C. R. Marotta to Karl Kniel, "Recriticality Potential of TMI-2 Core, U.S. Nuclear Regulatory Commission, May 14, 1979.

17 C. R. Marotta, "Recriticality Potential of TMI-2 Core," Transactions American Nuclear Society, 35, 272-274, November, 1980.

Appendix B

Total Preferential Failure of Batch 3

Hypothetically assuming a complete preferential collapse of all Batch 3 fuel, an analysis was performed to determine the boron concentration necessary to provide a 1.0% $\Delta\rho$ margin above the shutdown margin (K_{eff} = .98). The fuel and moderator parameters were optimized to provide the most reactive combination in a configuration of spherical segments in the bottom of the reactor vessel (see Figure 4-2).

The results in comparison to those in Section 4.0 are as follows:

Batch 3 Collapse Model	K_{eff}	Calculated ppm	System ppm	Fuel Assemblies	Pellet Size	Fuel VF
Maximum Credible	.98	3500	3700	19	1/4 radius	.55
Total	.98	4840	5040	60	whole	.63

APPENDIX B

TMI-2 DECAY HEAT REMOVAL ANALYSIS

APRIL 1982

PREPARED FOR

GPU NUCLEAR CORPORATION

BY

BABCOCK & WILCOX

1. INTRODUCTION

The Three Mile Island Unit 2 (TMI-2) reactor is currently in a decay heat natural circulation cooling mode. This report assesses the thermal status of the core and predicts the thermal response of the system to partial draindown of the reactor coolant system (RCS).

The criterion used in this study was based on TMI-2 operating procedures for natural cooling, which restricts the average incore coolant temperature to less than 170 F. This criterion was adopted as a conservative value for the recovery program to maintain a positive margin to boiling.

The analyses were directed at evaluating two concerns: (1) system effects of lowering the RCS level, and (2) coolant temperatures with lowered water level. In order to perform this evaluation it was necessary first to assess the current method of heat removal thereby establishing a basis for the predicted response to a lowered water level.

2. DECAY HEAT GENERATION RATE

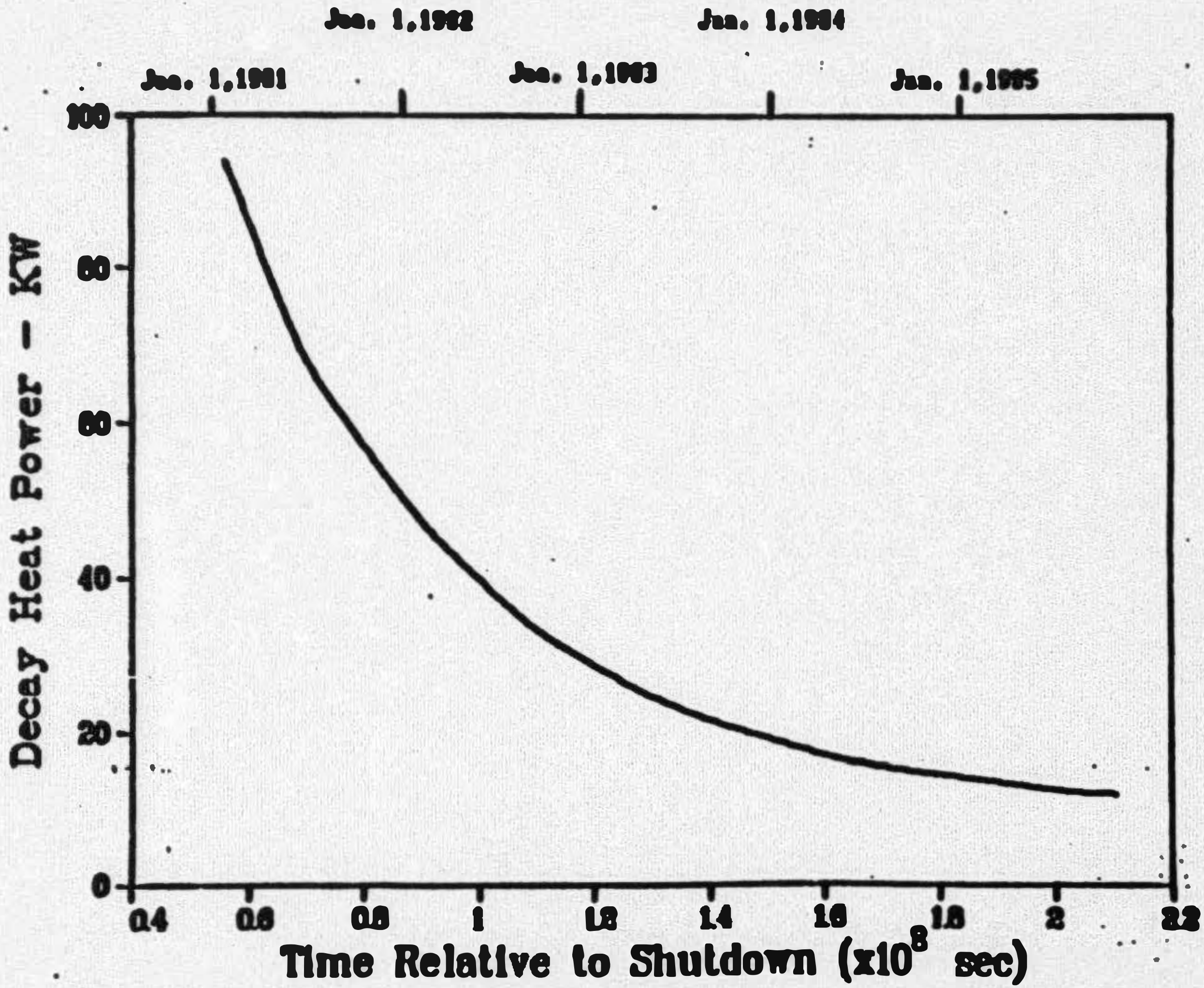
The decay heat power for TMI-2 was calculated. Figure 1 gives the decay heat power values.

The methodology used in the calculation of these decay heat power values is that prescribed in the ANSI/ANS 5.1-1979 standard. This standard provides the basis for determining the fission product decay heat power following shutdown of light water reactors. The decay heat is defined in terms of power as the summation of 3 sets (one for each fissioning species) of 23 energy-release-per fission terms (each prescribed in terms of operating time, shutdown time, growth rate and decay rate parameters). The decay heat power calculated is multiplied by an "upper bound term" for conservatism. The upper bound term is a function of shutdown time and varies from a factor of 1.0 at shutdown to a maximum of 1.6 at 10^8 second to 1.007 at 10^9 second. The derivation of this factor is based on an assumption that the reactor has operated for 4 years with an average neutron flux of 1.75×10^{14} n/cm^2/s (thermal), 3×10^{14} n/cm^2/s (epithermal). Since TMI-2 had only about 88 days of full power operation, these ANSI/ANS factors yield conservative results in this application.

The uncertainty in the method for calculating the upper bound decay heat value is estimated in the standard to be ±4%.

The decay heat power from activation products in the reactor materials is not included in the ANSI/ANS calculation. The decay heat power from ^{239}U and ^{239}Np are treated separately in the standard and are neglected in this analysis. This introduces no appreciable error in the decay heat values after 20 days shutdown time.

Figure 1
TMI-2 Decay Heat Curve



3. THERMAL ANALYSIS

3.1 Present Core Thermal Status

The decay heat currently being produced by the TMI-2 core, three years after shutdown, is approximately 45 kw. This relatively small power is being removed with no forced primary system flow from reactor coolant pumps or decay heat pumps, while the steam generators are in wet layup with no secondary side circulation. The measured temperature data for the primary system suggest that heat removal is occurring via conduction through the walls of the vessel, piping and steam generators, and is ultimately convected to the reactor building ambient air. The data also support the theory that some recirculation of fluid is taking place within the reactor vessel while some net migration of fluid around the RCS loops is also occurring.

To verify this theory of the current mode of heat removal a simple heat balance was established for the RCS. Using measured temperatures around the RCS, and calculated heat transfer coefficients for each component, the predicted heat removal can be determined. This heat removal is then compared to the predicted core heat generation for the particular date the temperature data were taken.

The results of this heat balance are summarized in Table 1. Due to changes in the reactor building water level over the time period represented by the data, the relative heat removal predicted from the various components changes for each date used for the data comparisons. Figures 2 through 5 provide the detailed heat balance data for each date and corresponding reactor building water level. Temperature data are shown on the RCS schematic. The incremental heat removal is calculated using the temperature data, the surface areas detailed in the attached tables and the overall heat transfer coefficient, U, also provided in the tables.

The heat transfer coefficients were determined using literature values for both the water and air side film coefficients. These film coefficients, empirically based, reflect the vertical or horizontal orientation of the component's surface and the shape of the component. The film coefficients neglect the presence of the mirror insulation present on all major RCS components. This insulation is effective at reactor operating temperatures due to its radiation heat transfer properties only; at lower temperatures where radiation heat transfer is negligible its insulating properties are assumed to be minimal. The validity of this assumption can be determined by comparing the results of the heat balance benchmark with the calculated decay heat values.

The components of the overall heat transfer coefficients are shown in Table 2. For each RCS component the conduction and convective coefficients are detailed along with the overall heat transfer coefficient. Literature sources for these coefficients are also provided with Table 2.

The results of the benchmark heat balance support the validity of the assumptions for the current mode of heat removal. Agreement between the predicted heat removal and the calculated decay heat production was obtained for the five configurations tested. Although some uncertainty exists relative to the actual decay heat production and the calculated heat losses (uncertainty in measured temperatures and calculated heat transfer coefficients), the agreement between heat production and heat loss supports the assumption of fluid recirculation in the vessel and net migration around the loop. Thus, the application of the heat balance technique and the corresponding heat transfer coefficients for analyzing other coolant levels appears reasonable.

Although agreement was obtained between the predicted heat loss and calculated heat production a number of uncertainties exist in the analysis. As noted in Section 2 the predicted decay heat values are subject to some uncertainty but should be conservative (slightly higher than actual).

The overall heat transfer coefficients shown in Table 2 are dominated by the air side convective film coefficient. These film coefficients are not precisely known. Also, the effect of the mirror insulation on the effective film coefficient is not known. However, the benchmark calculations were performed with a consistent application of the air side film coefficients although the proportions of heat transferred through the air interface were different for each of the cases. The different configurations (amount of RCS in contact with reactor building water) calculated suggest that these coefficients are reasonably accurate.

Other uncertainties present in the analysis include, (1) the use of average coolant temperature for a component rather than actual gradient, and, (2) the measurement uncertainty in the incore and loop temperatures themselves. These uncertainties will also exist for the partially drained configuration, and will be accounted for in the recommendations of Section 3.4 and Section 4.0.

The calculations are also subject to uncertainties which will not affect the analysis of the partially drained configuration. Heat removal via conduction from RCS components directly to reactor building water represents a significant portion of the total heat removal. The actual water level in the reactor building for each case of the calculations is approximate. However, the water no longer contacts any portion of the RCS, therefore, any uncertainty in this parameter will not affect the analysis of the partially drained configuration. The actual heat removal capacity of the reactor head and Control Rod Drive Mechanisms (CRDM) is unknown due to the likely presence of non-condensible gases in the RCS. It is likely that these gases partially insulate the head and CRDMs from the RCS water inventory thereby significantly reducing the heat transfer. Therefore, heat losses through the CRDMs were neglected for the benchmarking calculations. However, in the partially drained configuration the head and CRDMs will also be effectively eliminated as a source of heat removal since the RCS level will be low enough (1 ft above plenum cover) to prevent direct contact. Some heat removal via convection or evaporation may occur; however, this additional heat transfer is neglected as a conservatism.

The effects of the various uncertainties in the analysis cannot be accurately quantified. Rather than place arbitrary uncertainty and conservatism on the the analysis method, the recommendations based on these analyses reflect a qualitative judgment of the effect of these uncertainties.

Table 1. Results of Heat Balance

Date	Calculated heat loss	Estimated decay heat level
Sept. 26, 1981	54.9 kW	58.0 kW
Nov. 11, 1981	44.4 kW	52.0 kW
Jan. 1, 1982	49.0 kW	49.5 kW
Feb. 18, 1982	45.0 kW	46.5 kW

Table 2

Node	Water Side Film Coeff.[1-6] BTU/hr-ft²-°F	Wall Thermal Conductivity[7] BTU-in/hr-ft²-°F	Air Side Film Coeff.*[8] BTU/hr-ft²-°F	Overall Heat Transfer Coeff. BTU/hr-ft²-°F
A	166.2	370	.91	.76
B	94.7	370	.48	.48
C	94.7	370	.56	.54
D	24.9	370	.61	.59
E	37.1	370	***	3.16
F	37.1	370	.26	.25
G	64.7	370	.74	.70

***Portion of Steam Generator and Hot Leg Piping under water. Film coefficient of water on outside surface of mirror insulation is 31.7 BTU/hr-ft^2-$^\circ$F. The heat transfer coefficient for the water and mirror insulation combined was determined experimentally at Diamond Power to be approximately 13.0 BTU-in/hr-ft^2-$^\circ$F.

*The air side film coefficient is the most significant value in determining the overall heat transfer coefficient (U). The water side film coefficient and wall thermal conductivities have little effect on the overall U. The film coefficient of air for the different nodes was calculated using the equations which matched most closely the geometries of the different nodes.

Node		Applicable Geometry
A	-	Flat plate; horizontal
B	-	Cylinder; vertical
C	-	Cylinder; horizontal
D	-	Flat plate; vertical
E	-	Cylinder; horizontal
F	-	Cylinder; vertical
G	-	Flat plates; horizontal and vertical sections

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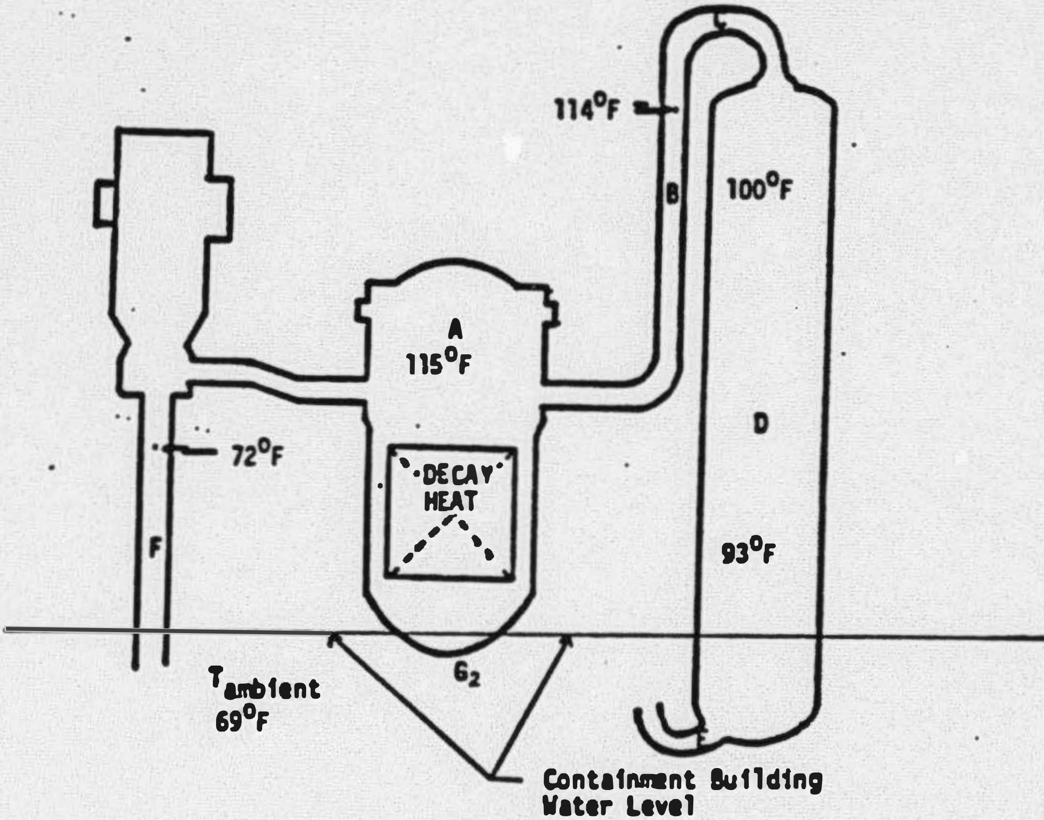
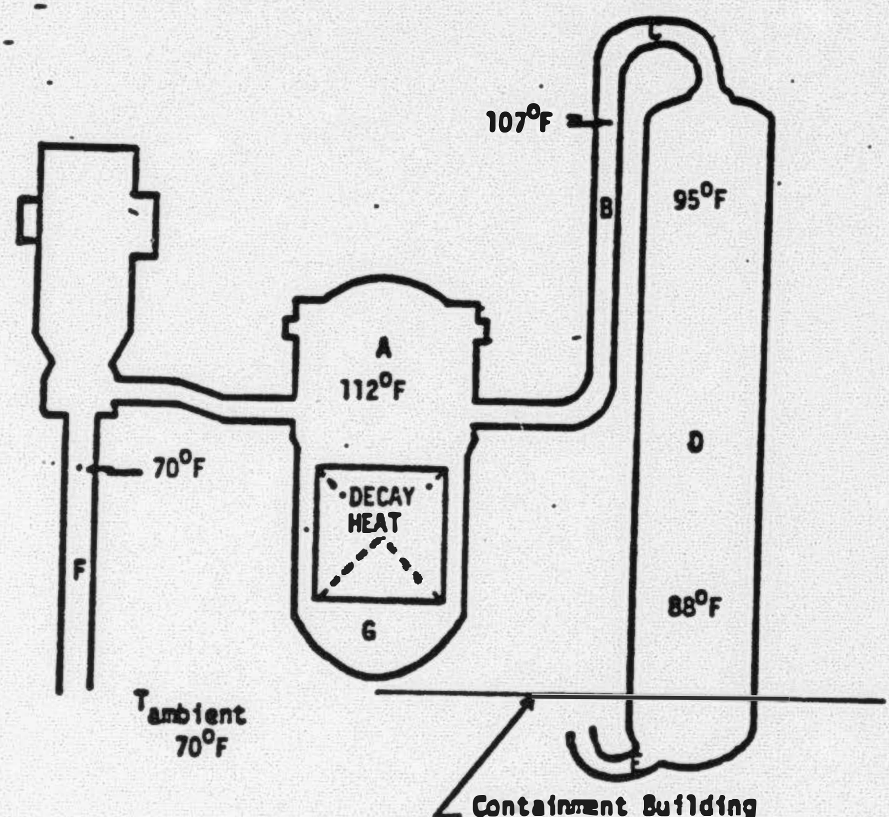
Figure 2. Losses Through RCS Loops,
September 26, 1981



$q = UA\Delta T$

Node	U, Btu/h-ft²-F	A, ft²	q, kW	Heat Sink
A	0.76	502	5.77	Ambient air
B	0.48	452	5.46	Ambient air
C	0.54	692	8.56	Ambient air
D	0.59	2056	19.45	Ambient air
E	3.16	361	8.02	Containment bldg. water
F	0.25	814	0.24	Ambient air
G_1	0.70	1135	6.99	Ambient air
G_2	3.16	15	0.41	Containment bldg. water
			54.9	

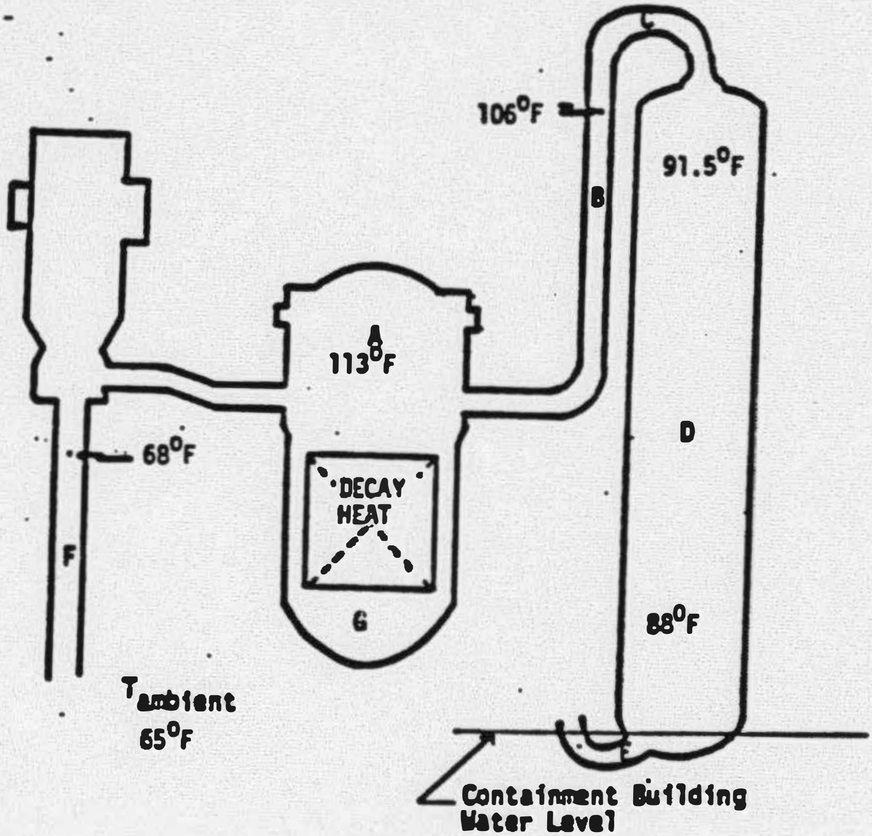
Figure 3.. Losses Through RCS Loops, November 11, 1981



$$q = UA\Delta T$$

Node	U, Btu/h-ft²-F	A, ft²	q, kW	Heat Sink
A	0.76	562	5.27	Ambient air
B	0.48	452	5.02	Ambient air
C	0.54	692	6.85	Ambient air
D	0.59	2070	14.66	Ambient air
E	3.16	337	5.55	Containment bldg. water
F	0.25	814	0.0	Ambient air
G	0.70	1150	7.03	Ambient air
			44.4	

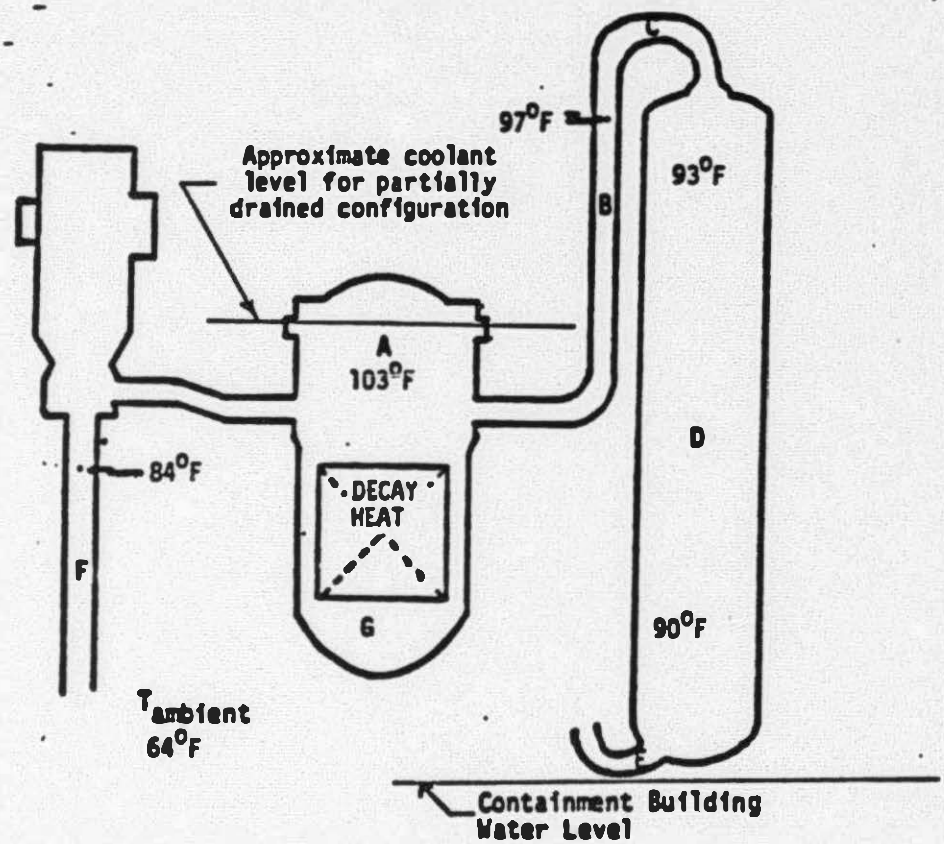
Figure 4. Losses Through RCS Loops, January 1, 1982



$$q = UA\Delta T$$

Node	U, Btu/h-ft^2-F	A, ft^2	q, kW	Heat Sink
A	0.76	562	6.03	Ambient air
B	0.48	452	5.61	Ambient air
C	0.54	692	7.05	Ambient air
D	0.59	2102	15.03	Ambient air
E	3.16	233	7.50	Containment bldg. water
F	0.25	816	0.20	Ambient air
G	0.70	1150	7.62	Ambient air
			49.0	

Figure 5. Losses Through RCS Loops,
February 18, 1982



$q = UA\Delta T$

Node	U, Btu/h-ft²-F	A, ft²	q, kW	Heat Sink
A	0.76	562	4.91	Ambient air
B	0.48	452	4.46	Ambient air
C	0.54	692	6.87	Ambient air
D	0.59	1738	15.[illegible]5	Ambient air
E	0.59	419	2.18	Ambient air
F	0.25	1050	3.08	Ambient air
G	0.70	1150	7.50	Ambient air
			45.0	

3.2 Radial Core Coolant Temperature Profile

The TMI-2 inspection program currently planned as part of the overall recovery program will be performed with the TMI-2 operating procedures. For the partial draining of the RCS, TMI-2 operating procedure 2102-3.4, Revision 2, "Reactor Coolant System Operation with Core Cooling via Natural Heat Loss," states that the average incore thermocouple temperature must be less than 170 F and all operable thermocouple temperature readings must be less than 210 F.

The basis for satisfying this criterion with respect to the average temperature was established in Section 3.1 and is discussed further in Sections 3.4 and 4.0. Predicting the maximum thermocouple response based on the average temperature, however, requires an evaluation of the maximum/average temperature history.

The relationship of maximum to average incore thermocouple values was evaluated through the use of the core coolant temperature map shown in Figure 6. This map constructed from data taken October 27, 1981, has an average temperature of 113 F. Most values are grouped in the range of 110 F to 120 F; total range of values is 90 F to 143 F. The difference between the maximum and average temperatures of 30 F is indicative of values observed over the preceeding six month time period (September 1981 to February 1982). This difference between the average and maximum values is expected to decrease slowly with the decrease in decay heat level. Thus, it is likely that maintaining the average thermocouple reading below 170 F will ensure that the maximum thermocouple value will not exceed 210 F. However, individual thermocouples could be subject to random error such that the 40 F delta between maximum and average thermocouple readings is exceeded. While it is reasonable to interpret the present map as including such errors, it is impossible to predict this behavior precisely. Therefore, the full incore thermocouple temperature map should be monitored closely following the partial draining of the RCS. These data can be used to identify obviously erroneous thermocouple readings and eliminate these readings from consideration in meeting the 210 F maximum temperature criterion.

Figure 6

Incore Thermocouple Temperatures
October 27, 1981

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A															
B							114	114							
C						115			114	111			113		
D					120					108				111	
E				131			117		118						
F			132				106	98				115	102		
G		117				112			118		121		113		
H	113				116			143					91		
K					111						120				
L		116	119												
M			127				113		96	117				111	
N				114				104	101						
O					116	90				115			108		
P						105									
R							114			115					

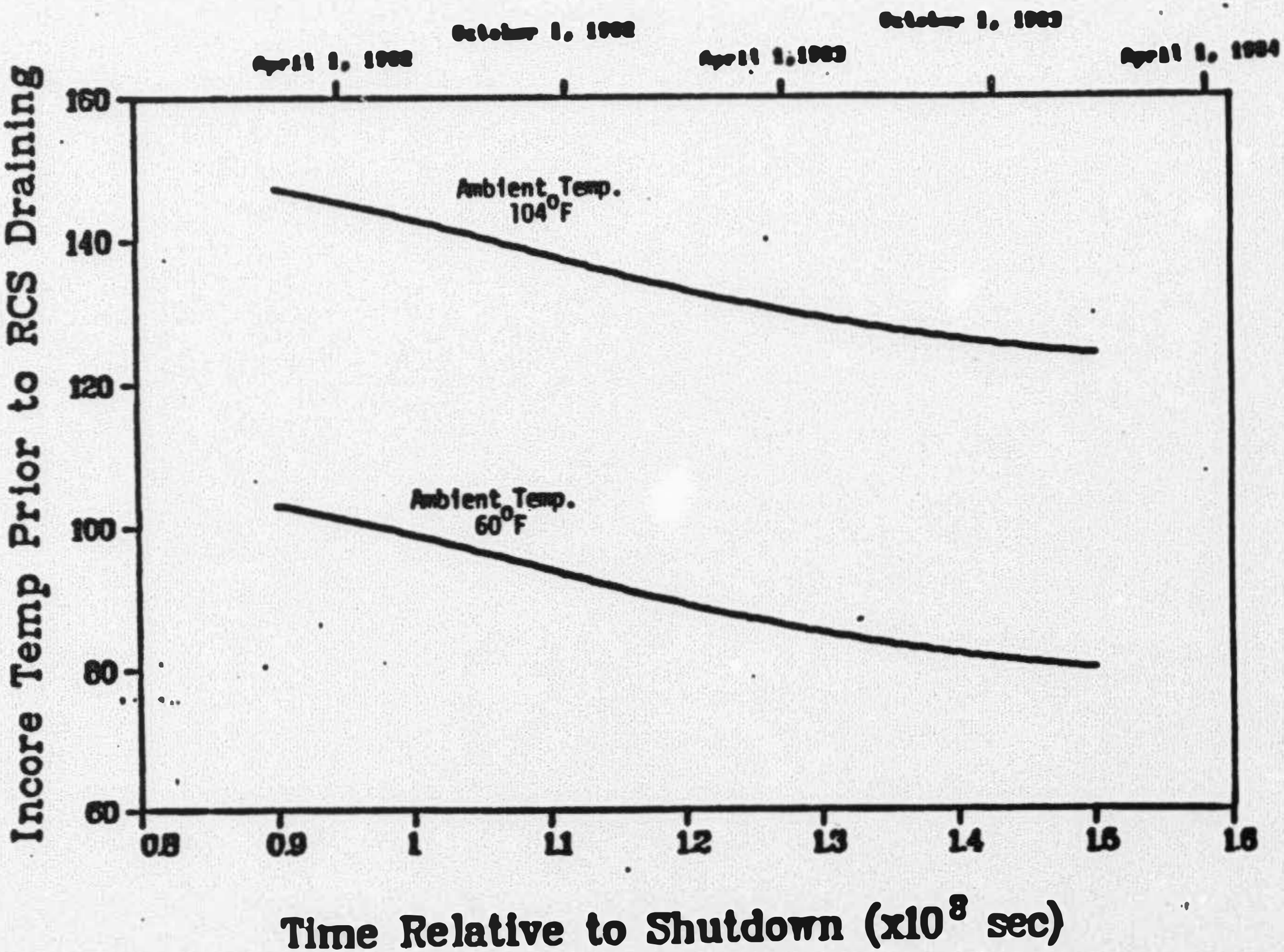
3.3 Core Thermal Status Before Pre-Head Lift Inspection

The analysis method established in Section 3.1 was utilized to predict the incore temperature prior to partial draining of the RCS. With no external cooling mechanism in effect, the incore temperatures will depend on the amount of heat being produced and the temperature of the heat sink (reactor building ambient air). Thus, the prediction of core average temperature in Figure 7 is a function of both the time after shutdown (decay heat decreasing with time) and the reactor building ambient air temperature. Incore temperature predictions are made for the time period of January, 1982, through January, 1984, assuming no contact between the RCS and the reactor building water inventory.

Measured incore temperatures from September, 1981, to February, 1982, have remained relatively constant. (See Figures 2-5) Although decay heat production decreased during this time, the reactor building water level was also being lowered. As this level decreases, the heat transfer from the RCS is reduced (reactor building water in contact with the RCS provides improved heat transfer relative to air). As of February 3, 1982, the reactor building water level no longer contacted any portion of the RCS. The data reported in Figure 5 (February 18, 1982) are consistent with this lowered water level.

The curves of Figure 7 demonstrate that the average incore temperature should remain below 170 F indefinitely in the current cooling mode as long as the present coolant inventory is maintained. It can also be observed that the predicted incore temperature is sensitive to the reactor building ambient air temperature. The relationship between incore temperature and ambient air temperature is essentially one-to-one, i.e. each $1^{O}F$ change in the ambient air results in a $1^{O}F$ change in the core coolant temperature.

Figure 7
Incore Temperature vs Time After Shutdown
Before RCS Draining



3.4 Decay Heat Removal With a Partially Drained RCS

The recovery procedures at TMI-2 require the partial draining of the RCS to perform inspections prior to head removal. The incore coolant temperature response to this partial draining has been evaluated based on the method established and benchmarked in Section 3.1.

Heat removal in the partially drained configuration is assumed to occur only through the reactor vessel walls and lower head. Based on an anticipated coolant level one foot above the plenum cover (see Figure 5), the upper head could be effectively insulated from the coolant inventory. Although some heat loss through evaporation at the liquid surface and condensation on the vessel head would occur, this particular mechanism is conservatively neglected. Also not considered in this evaluation are heat losses which could occur in the RCS cold leg piping. In the partially drained configuration no flow path around the loop remains. Thus, with no net mass of fluid moving around the loops it is conservatively assumed that a negligible amount of heat could be lost via conduction through the stagnant water in the cold leg piping.

Based on this assumed heat transfer geometry the incore coolant temperatures are found to depend directly on the core power (decay heat) and the temperature of the heat sink (reactor building ambient air). Thus, the predicted incore coolant temperature is shown in Figure 8 as a function of time after shutdown and the reactor building ambient air temperature. From this figure it can be observed that the average coolant temperature may exceed the 170 F criterion depending on the date of partial draining and the actual reactor building air temperature. However, it should be noted that the temperatures displayed in Figure 8 are steady state values. The mass of metal and coolant in the reactor vessel is significant relative to the small amount of power currently being produced. To quantify the magnitude of the time delay that would be expected in reaching the temperatures predicted in Figure 8, the core heat up rate was determined.

The core coolant heatup rate after partial draining of the RCS is shown in Figure 9 as a function of time after shutdown and reactor building ambient air temperature. This curve is based on assumed coolant temperature of 170 F; at coolant temperatures below this value heatup rates would be slightly higher. The heat capacity of the water in the reactor vessel (to 1 foot above the plenum cover) is the dominant effect in the analysis although the heat capacity of fuel and metal components (vessel and internals) was also included. As in the analysis of Figure 8, the coolant inventory contained in the hot and cold legs is not considered in the heatup rate calculations.

The time response shown in Figure 9 is significant relative to the predicted temperatures of Figure 8. Whereas the curves of Figure 8 represent steady state values, the results of Figure 9 indicate that the core could take 2 to 4 weeks to reach 170 F from an initial temperature of 100 F-110 F (reactor building air temperature of 104 F, date of draining - late 1982). Thus, the results of Figure 8 should be taken as guidelines for the potential need for supplemental cooling rather than absolute requirements.

The potential need for supplemental cooling is shown in the predicted temperatures of Figure 8. Guidelines for the amount of supplemental cooling which could be required are presented in Figure 10. In order to maintain the predicted incore temperature below 170 F, the required supplemental heat removal is shown as a function of time after shutdown (decay heat decreasing with time) and reactor building ambient air temperature. The excess heat which must be removed to maintain the predicted incore temperatures below 170 F is shown in Figure 10 to be quite small (on the order of 5-10 kw for a mid-1982 draindown with an ambient air temperature of ~80 F).

As shown in Figure 10 the reactor building ambient air temperature significantly affects the supplemental cooling requirements. Assuming no supplemental cooling the maximum building air temperature to maintain predicted incore temperatures below 170 F was determined. This maximum building air temperature is shown in Figure 11 as a function of the time after shutdown at which partial draining of the RCS is performed.

As in the case of Figures 8 and 10, Figure 11 provides guidelines for operation. Due to the uncertainties identified in Section 3.1, these figures should be used as guidelines rather than requirements. Due to the low heatup rates involved (Fiugre 9) a significant amount of time will be available to alter the core cooling mechanism or restore the coolant level, if required.

Figure 6
INCORE TEMPERATURE AFTER RCS DRAINING WITHOUT SUPPLEMENTAL COOLING

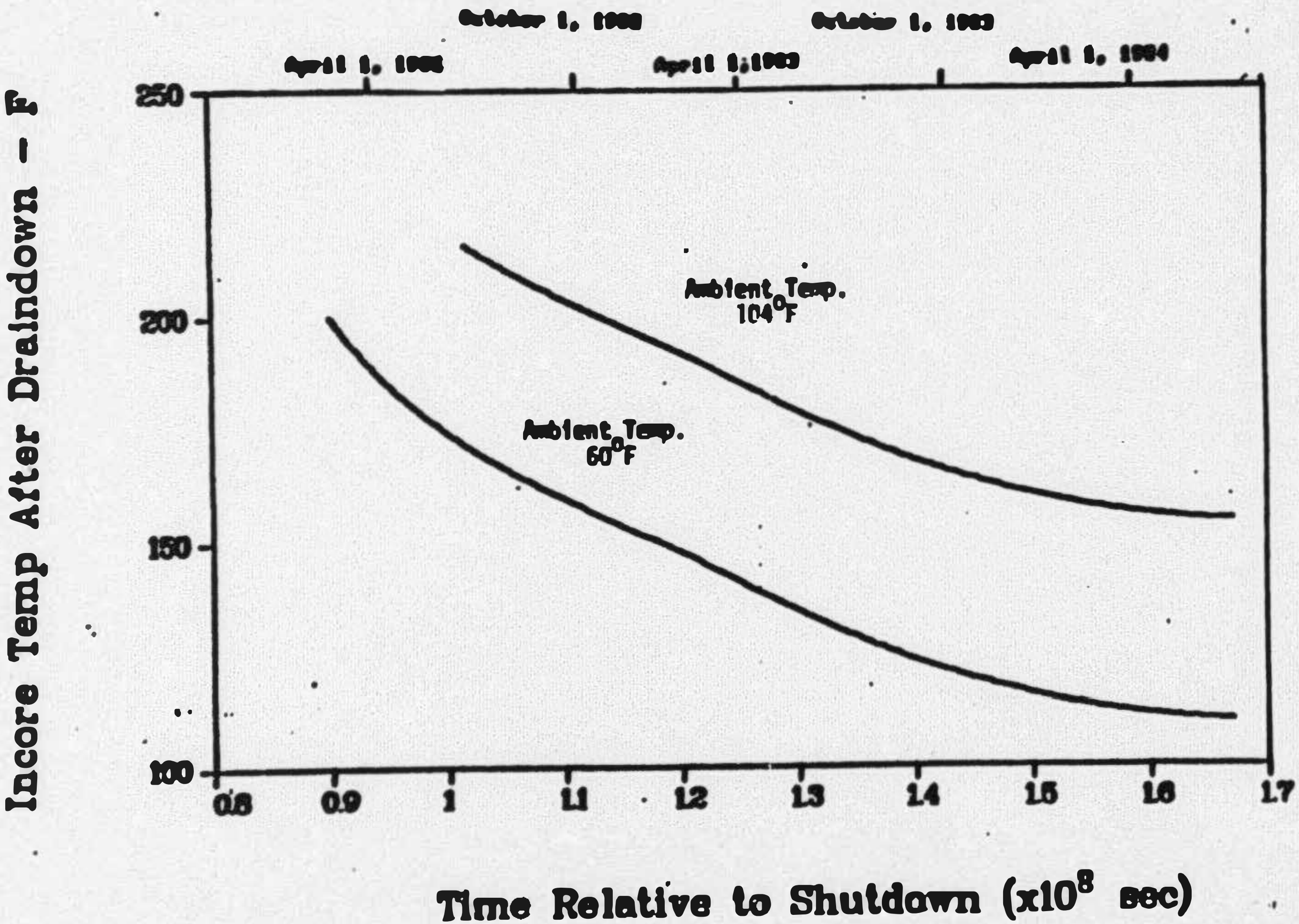


Figure 9
Core Heat-up Rate After RCS Draining

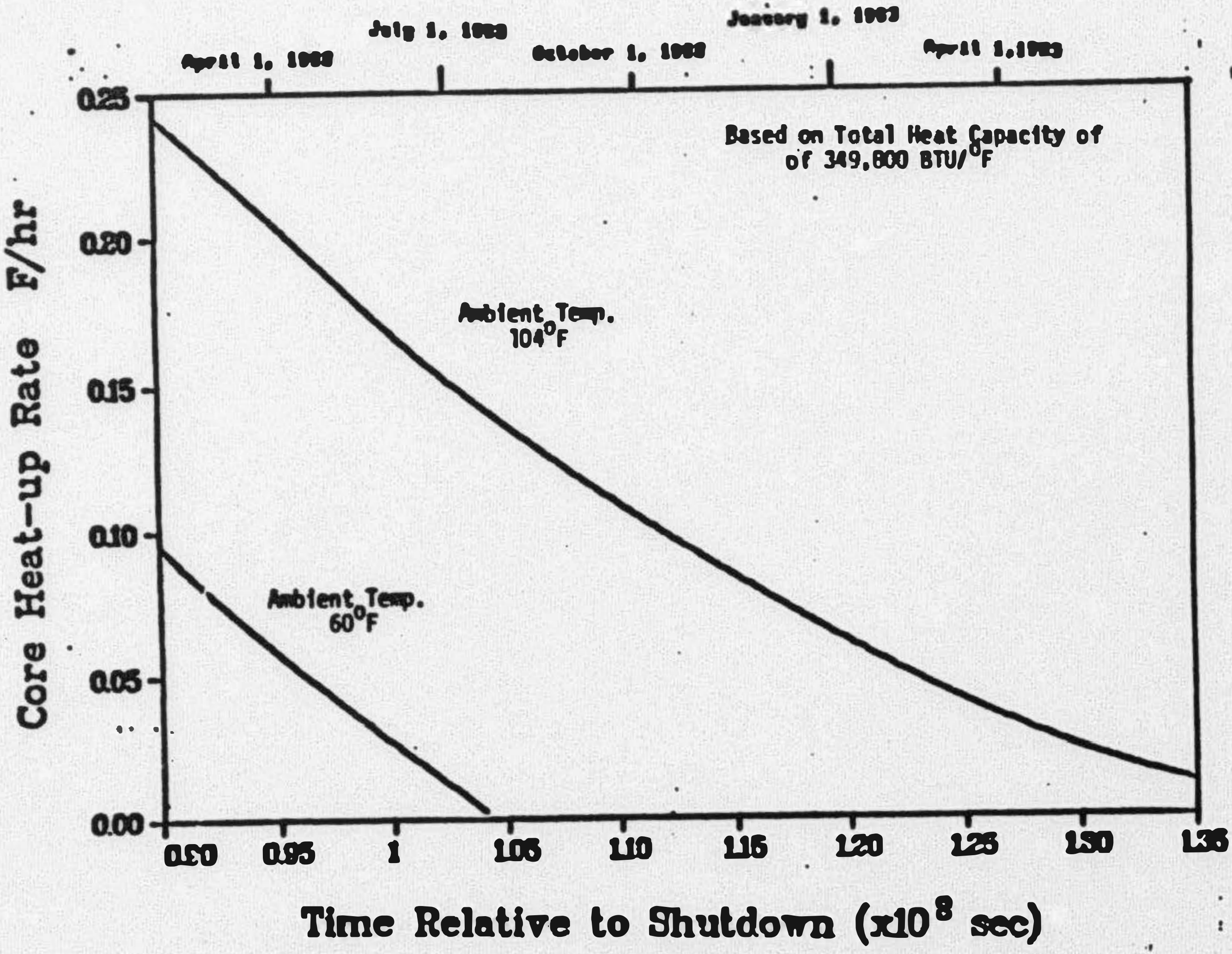


Figure 10

ADDITIONAL HEAT REMOVAL AFTER RCS DRAINING FOR MAINTAINING CORE AVG TEMP AT OR BELOW 170 F

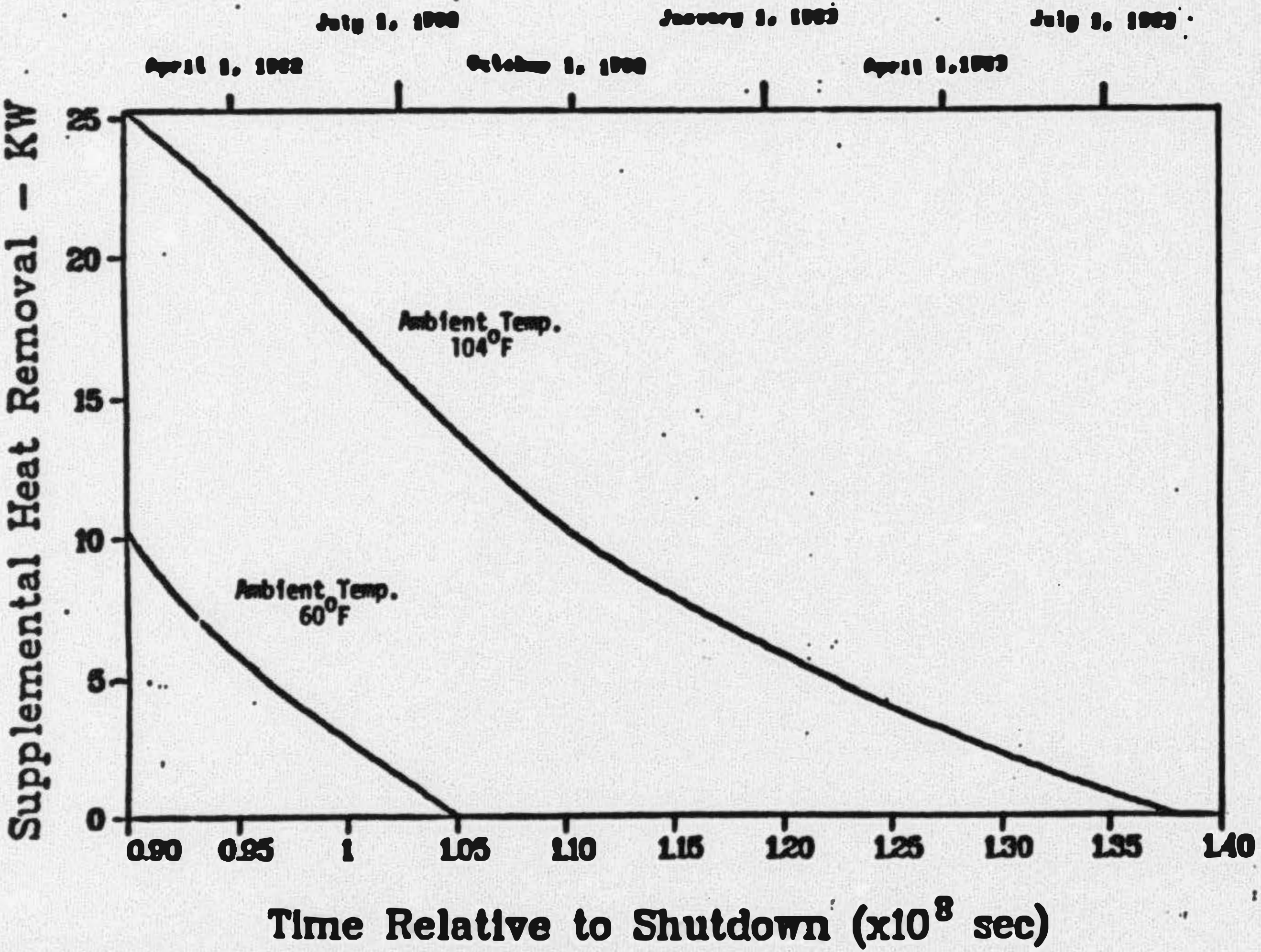
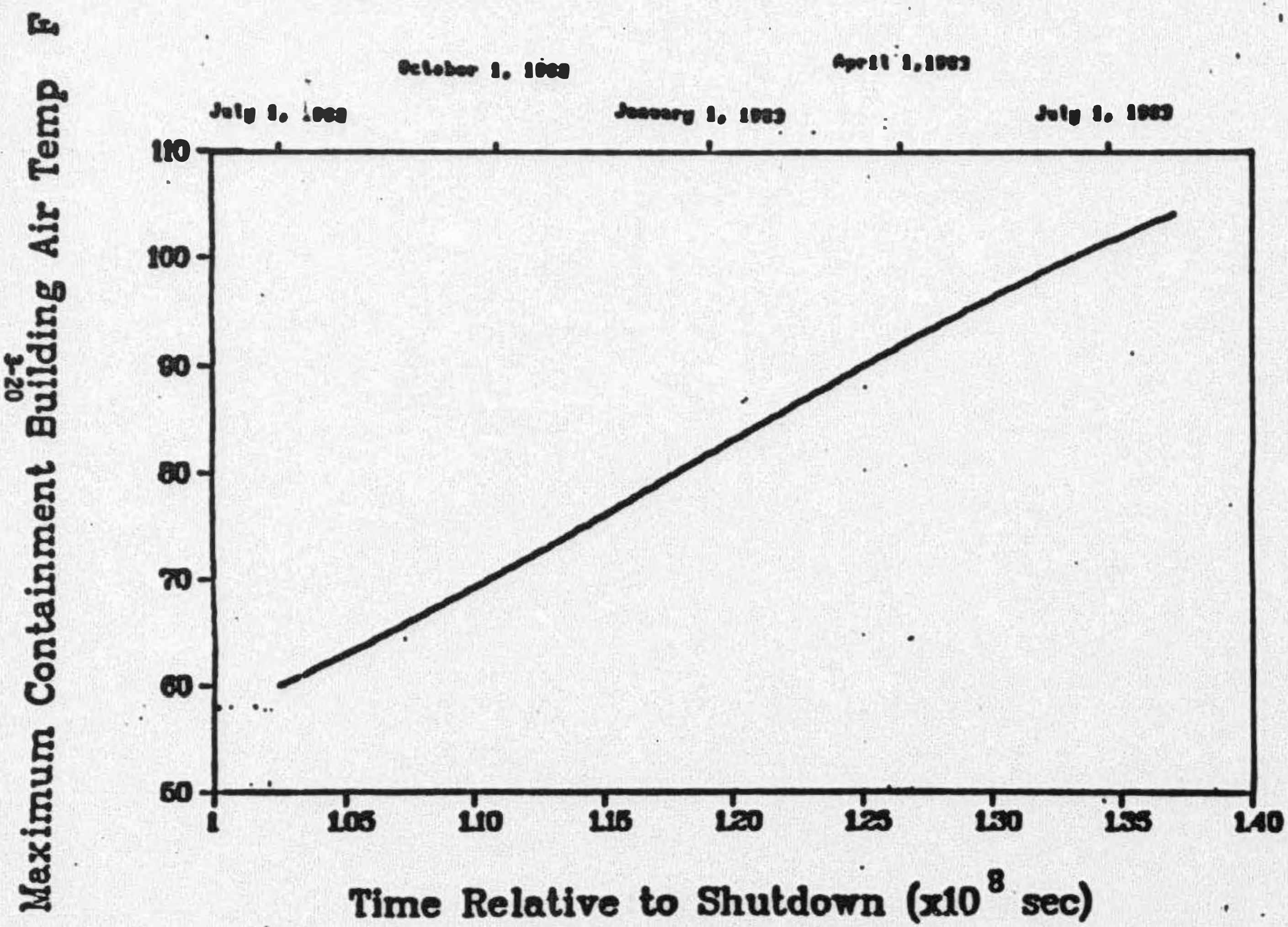


Figure 11

CONTAINMENT BUILDING AIR TEMP REQUIRED AFTER RCS DRAINING TO KEEP INCORE TEMPS BELOW 170 F



4. SAFETY EVALUATION

The losses-to-ambient core cooling mode for a partially drained RCS has been evaluated with respect to the safe operation of the plant. The basis for evaluating the adequacy of this cooling mode is taken from TMI-2 operating procedure 2102-3.4, Revision 2, "Reactor Coolant System Operation with Core Cooling via Natural Heat Loss." This procedure states that the average incore thermocouple temperature must be less than 170 F and all operable thermocouple temperature readings must be less than 210 F.

Guidelines for the predicted average incore temperature after partial draining of the RCS are presented in Figures 8, 10 and 11. These guidelines indicate that under certain conditions the heatup of the reactor coolant could ultimately approach the 170 F criteria if measures were not taken to improve heat removal. Due to the uncertainties identified in Section 3.1 it is recommended that some means of improved heat removal be provided in the procedure as a back-up position. Core heatup rates shown in Figure 9 indicate that a significant time period (days to weeks) will be available to implement the improved heat removal, if required.

Conservatisms were introduced into the analyses through the elimination of all heat removal from the RCS via the reactor vessel head and CRDMs, and cold leg piping. Therefore no supplemental heat removal may actually be required.

The maximum thermocouple reading is expected to remain below 210 F by maintaining the average thermocouple reading below 170 F. This conclusion is based on the core radial thermocouple map of Section 3.2 (Figure 6) and on the expected trend in maximum/average readings with decreasing decay heat production.

Containment building ambient air temperature has been shown to significantly affect the cooling requirements in the partially drained configuration. Maintaining the ambient air temperature at the required level (as per the guidelines of Figure 11) is an effective means of holding average incore temperatures below 170 F without requiring supplemental heat removal. With the very slow

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An alternate means of providing supplemental heat removal would be to utilize air flow across the reactor coolant inside the vessel head. Such a flow path could be established between CRDM and reactor vessel head thermocouple penetrations. Maintaining a flow across the coolant in the partially drained configuration would provide direct convective cooling and evaporative cooling. With the slow heatup rates involved, the adequacy of this cooling mechanism could be verified by monitoring the actual incore thermocouple temperature response.

Supplemental cooling could also be accomplished by adding water to the RCS by means of the Decay Heat Removal, Mini Decay Heat Removal, or Standby Pressure Control System.

APPENDIX C

REACTOR COOLANT SYSTEM (RCS) DILUTION SAFETY EVALUATION

I. INTRODUCTION

During the "Quick Look" testing, reactor shutdown (subcriticality) will be assured by the presence of boron in the reactor coolant. The Quick Look Safety Evaluation has shown that maintaining RCS boron concentrations of 3500 ppm or greater assures subcriticality under all credible conditions.

TMI Unit 2 operations during the past two years has demonstrated that it is possible to maintain a controlled boron concentration in the RCS. However, during the "Quick Look" testing the RCS conditions will differ from those that existed during the previous two years. The primary coolant level will be lowered and the primary coolant pressure will be reduced. 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 appendix is to review the precautions that will be taken to assure that the required RCS boron concentraton will be maintained.

The Reactor Coolant System temperature and chemistry will not be significantly affected by the Quick Look test and, hence, boron solubility will remain essentially unchanged. The only way RCS boron concentration can be changed in an uncontrolled manner during this test is by dilution of the RCS coolant with water that is either, unborated or borated below 3500 ppm. This discussion will therefore review the methods that will be used to prevent boron dilution.

Uncontrolled boron dilution will be prevented by a combination of prevention, monitoring, and corrective actions.

The following discussion shows that the procedures in effect during the time 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 shutdown.

II. 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 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 to assure that they will not be creditable sources of boron dilution.

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.

1. Steam Generator

One potential source of dilution of the RCS boron is steam generator 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 coolant pressure, any leakage would be from the RCS to the secondary system.

During the "Quick Look" the primary system pressure will be reduced to atmospheric pressure. To preclude RCS dilution while the RCS pressure is reduced, procedures require that water levels in the secondary side of the steam generators will be maintained lower than the levels in the primary side. In this manner the preferred potential leakage path, from primary to secondary volumes, is maintained. Under these conditions the secondary volume of the steam generators will not be a credible source of RCS boron dilution.

2. Makeup and Purification/Standby Pressure Control Systems

These systems are borated greater than 3600 ppm and will be operated by approved "Quick Look" procedures to supply make-up water to the Reactor Coolant System whenever leakage necessitates addition. The Make-up pumps, MU-P-1A/B/C will be tagged "off" and portions or connections to these systems that are not used for make-up will be isolated. In addition, the emergency procedure uses portions of the Waste Decay Liquid system to rapidly refill the Reactor Coolant System from the RC bleed tank via the waste pump, WDL-P-5 A/B. To preclude any inadvertent operation of this emergency injection, isolation valves, MU-V9 and MU-V10 will be tagged closed.

3. 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.

4. Submerged Demineralizer System

During the "Quick Look" evolution the SDS may be operated to process the present supply of unprocessed water. This evolution will not create a dilution concern since the SDS will be isolated from the Reactor Coolant System by tagging closed the injection valves, MU-V9 and MU-V10.

5. Other Systems

TABLE 1

Decay Heat Removal
Mini Decay Heat System
Core Flood System
Intermediate Closed Cooling Water
Nuclear Services 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 systems in Table 1 of coolant containing boron concentrations less than 3500 ppm to the RCS.

A. The systems in Table 1 have been reviewed and isolation valves in the flow paths have been tagged shut.

B. 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 II 2, 3, and 4). The position of these valves will be confirmed every 24 hours during this period.

C. All pumps in these systems; except nuclear services closed cooling water which cools the instrument air compressors and waste gas compressors; will be tagged out to further preclude the inadvertent transfer of coolant to the RCS.

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

E. The systems in Table 1 will not be operated in any manner, except for emergencies where plant safety is involved, while the control drive mechanisms are open during the Quick Look. All other times their operation will be limited to those actions necessary to support plant operation. (These essential needs will not include additions to the RCS. Such action will only be done for plant safety or when approved by PORC on a case by case basis.) Such operation will require the written approval of the Shift foremen and will be conducted under his direct surveillance. During this period any change in status or operation of the systems in Table 1 will be entered into the control room log. Surveillance testing for the systems in Table 1 will be postponed during the time the RCS is at atmospheric pressure.

III. ACTIONS TAKEN TO DETECT AND TERMINATE INADVERTENT BORON DILUTION

The actions described in Section III 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.

The boron concentration will be monitored by monitoring the RCS coolant level. After the RCS level has been reduced, a base system leakage rate will be established. Using this leakage rate a plot of predicted level versus time will be given to the operations staff. Superimposed on this predicted level will an $\pm$ alarm and + action level limit. The alarm levels will be 24 inches above and below the base levels. The action level is 12 inches above the base level. If it is assumed that unborated water is being added to the RCS, the RCS boron concentration will still be above 3500 ppm at the higher alarm level. The lower alarm level is used as a precaution to indicate a possible increase in plant leakage and, hence, a need change the level.

Procedures will require the following action in the event the high action level is exceeded.

1. The control room operation log discussed in Section II 2E will be reviewed to attempt to determine the source of dilution.
2. The position of all isolation valves and status of all pumps discussed in Sections II 2B and C shall be checked.
3. Storage tank levels (Section II 2D) will be checked to determine the source of coolent dilution.
4. Quick Look operations shall be temporarily terminated and the mechanism seal shall be replaced if the high alarm level is reached.

If the above actions do not stop the increase of RCS coolant level and high alarm level is reached, the TMI Unit 2 Emergency Procedure 2202-1.2 "Unanticipated Boron Dilution" will be used to increase the RCS boron concentration.

In the event the lower alarm level is reached, a new base level curve will be established with appropriate alarm and action level limits.

In addition to monitoring the RCS coolant levels and the alarm and action levels discussed above, the source range neutron instrumentation will be monitored. A base count rate will be established after the coolant level has been lowered. An increase in count rate for more than one minute, of two times the base count rate shall be considered an alarm limit. An increase of five times the count rate for less than one minute shall also be considered an alarm limit. On reaching these limits the response will be the same as for the RCS coolant high level alarm.

V. SUMMARY

The actions discussed above are considered sufficient to preclude inadvertent boron dilution. In the unlikely event such dilution were to occur, procedures 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 considered credible.