Nuclear

GPU Nuclear
P.O. Box 480
Middletown, Pennsylvania 17057
717-944-7621
Writer's Direct Dial Number:

June 15, 1982
4400-82-L-0095

TMI Program Office
Attn: Mr. L. H. Barrett, Deputy Program Director
U. S. Nuclear Regulatory Commission
c/o Three Mile Island Nuclear Station
Middletown, Pennsylvania 17057

Dear Sir:

Three Mile Island Nuclear Station, Unit 2 (TMI-2)
Operating License No. DPR-73
Docket No. 50-320
APSR Insertion - Safety Evaluation Report (SER)

Attached, per our commitment in Technical Specification Change Request No. 37, is a copy of the Safety Evaluation Report for the APSR Insertion Test, currently scheduled for June 24, 1982. The SER is forwarded to you to provide additional information related to the safety of the planned test. The SER is also provided for you use in reviewing and approving the required procedures pursuant to Technical Specification 6.8.2. Your timely consideration of this document to suport the current schedule would be greatly appreciated. If you have any questions or desire further information, please contact me.

Sincerely,

J. J. Barton
Acting Director, TMI-2

JJB:RBS:djb

Attachment

cc: Dr. B. J. Sndyer, Program Director - TMI Program Office

U.S. NUCLEAR REGULATORY COMMISSION
1982 JUN 15 PM 9 04

D009

8206180152 820615
PDR ADOCK 05000320
P PDR

Safety Evaluation

FORMAT: The format of this safety evaluation is in accordance with Technical Functions Procedure EP-016. EP-016 is a procedure providing guidance for preparation of general safety evaluations. Accordingly, some portions of EP-016 are not applicable to this specific evaluation. Where this is the case, "Not Applicable" will be indicated.

1.0 TITLE:

SAFETY EVALUATION OF TEST PROCEDURE C-9631 (EG&G DCC 7076) "Dynamic In Situ Test of Axial Power Shaping Rod Drive Mechanisms".

2.0 PURPOSE:

Test Procedure C-9631 will be used to move, one at a time, the eight axial power shaping rod assemblies (APSR) in the TMI Unit 2 core. These assemblies are currently withdrawn about 35 inches or 25% of their full travel. The purpose of this test is two fold:

A. The ability to move some or all of the APSRs will provide an insight into the extent of core and upper plenum damage. This early insight will provide time to factor this information into plans for subsequent inspections, head and upper plenum removal and core removal.

B. Prior to head removal it is necessary to decouple the APSR lead screws. This can be most easily accomplished if the assemblies are fully inserted. Hence, inserting the APSRs into the core during this test is one step in the recovery process.

The procedure calls for withdrawing an assembly 3/16 of an inch and then inserting it fully into the core. Motion will be monitored by detecting pole slippage, either acoustically or electrically, and through the use of the individual position indicators if they are operational. Pole slippage occurs when the electrical field in the stator rotates but the mechanism rotor does not stay in synchronism, either lagging or remaining stationary. The portable service power supply will be used to run each mechanism, one at a time, from the control rod drive mechanism cabinets in the cable room. The mechanisms will be run without stator cooling water and stator temperature will be limited by monitoring stator thermocouples, if they are available, or limiting the time power is applied to the stator.

This test is supported by the results of two prior tests.

A. The first test constituted a static test of each APSR stator's electrical properties. Insulation and winding resistance, capacitance and inductance were measured from the terminal connections in the cable room. In addition, time domain reflectometry measurements were made on each stator. The results of these measurements indicate that the stators are electrically operational.

B. The second series of tests were performed by EG&G and Babcock & Wilcox at the Diamond Power Specialty Company Test Facility in Lancaster, Ohio. These tests:

1. Confirmed that lead screw motion can be monitored using pole slippage.

2. Confirmed that latching could be determined acoustically.

3. Measured mechanism forces transmitted to the lead screw as a function of stator power.

4. Measured stator heat-up rates without cooling water.

5. Measured lead screw motion under latching conditions.

6. Attempted to confirm the ability to use pole slippage as a means of determining the weight of a lead screw/control assembly (the experiment showed that this technique may not be successful).

7. Confirmed the ability to use the portable service power supply to provide the necessary experimental control.

8. Determined the effect water, or the absence of water, in the mechanism housing has on its performance.

Briefly, the test sequence for an individual APSR is as follows:

Using the (portable) service power supply, the latching current will be applied to a selected APSR mechanism. The motor windings to which latching current will be applied will be selected to minimize the possibility that the mechanism rotor will rotate during latching. Confirmation of latching will be obtained acoustically, by means of a pickup attached to the selected mechanism.

Operating in the single step mode, attempts will be made to move the control assembly a total of 3/16 inches outward. This will be accomplished in a total of six steps. During each step, electrical and acoustic outputs will be monitored for evidence of pole slipping, i.e., a stuck assembly. The current selected for this and subsequent operations will be varied, the force applied by the drive line (assuming it is initially stuck) will vary from about 500 pounds to a maximum of about 1400 pounds. After moving or attempting to move the assembly outward, inward motion will be attempted, again in the single step mode. A total of about twelve inward steps will then be performed, monitoring electrically and acoustically for evidence of pole slipping.

When the twelve-step inward motion is successfully completed, the assembly will be moved inward in the "jog" mode, again monitoring for evidence of pole slipping. Jog motion is a continuous drive to the assembly at three inches per minute, one-tenth normal "run" speed. Confirmation of motion should be obtained from the absolute position indication system, if it is operational. If the absolute position system proves inoperable, but no evidence of pole slipping is obtained, relative position, based on the number of steps applied to the mechanism, will be used to determine its position. The above referenced testing has shown that incremental rod motion occurs if pole slippage is not detected. When the assembly reaches the bottom of its travel, its position will be confirmed either by the absolute position indication or by evidence of pole slipping, or both.

Should sticking occur at any juncture of this sequence, controlled increases in force (current) will be made, up to the maximum force the power supply can provide, about 1400 pounds.

It is conceivable that, depending on the information obtained from the APSR tests, this testing may be extended to selected safety/regulating control rod assemblies. These tests would withdraw a control rod assembly about one half inch from its fully inserted position and then re-insert it. This would be done one assembly at a time. As the mechanisms are essentially the same as the APSR drive mechanisms, and the motions will be less than the full insertion of the APSR's (implying lower reactivity additions here than in the APSR case) this safety evaluation is considered to apply also to testing the safety/regulating control rod assemblies.

3.0 SYSTEMS AFFECTED:

3.1 Systems

1. APSR.
2. APSR Drives.
3. Control Rod Drive Portable Service Power Supply.
4. Control Rod Drive Position Indicator.
5. Control Rod Drive Stator Thermocouples.
6. Test Instrumentation.

3.2 Drawings

	TITLE	BURNS & ROE DRAWING NUMBER
1.	CRD Cable Chase Layout	3179 Sheet 1
2.	Stator Block Diagram	3024 Sheet 76
3.	External Connections for Cabinet 116	3326 Sheet 16A
4.	Block Diagram API Cable	3024 Sheets 61B & 61C
5.	External Connections for Cabinet 101 API Cable	3326 Sheets 1 & 1A
6.	Block Diagram, Iron Constantan Thermocouples	3024 Sheet 80
7.	External Connections for Terminal Box RI 139	3037 Sheet 117

3.3 Documents which Describe Affected Components:

3.3.1 Docket 50320 "Safety Analysis Report", Metropolitan Edison Co., June 1974

3.3.2 Technical Specifications:

A. Technical Specification Change Request No. 37

B. TMI Unit 2 Proposed Technical Specifications, Appendix A:

1. Section 3.1.1 Boron Control

2. Section 3.3.1 Neutron Monitoring Instrumentation

3. Section 3.6.1 Containment Integrity

4. Section 4.1.1.2 Boron Concentration

5. Section 4.3.1 Neutron Monitoring Instrumentation

3.3.3 Applicable System Descriptions:

A. Axial Power Shaping Control Rod Drive Mechanism Instruction Manual, Burns & Roe Document 43-53-005.

B. CRDM General Cable Layout, Burns & Roe Document 7-00-1306

C. Instrumentation (API) Cable Assembly, Burns & Roe Document 07-00-0811

D. Thermocouple Cable Assembly, Burns & Roe Document 7-00-1307

E. Test Guide TG 06 000 23 Jersey Central Power & Light Three Mile Island Unit 2 Physics Test Manual, Babcock & Wilcox Document NPGD-TM-229 (Contains proprietary information)

3.3.4 Applicable Drawings:

Shown above in Section 3.2.

3.3.5 Other References:

A. Test Results "Static in Situ Test Results of the APSR and Shim Safety Rod Drive Mechanisms," 007 007 099 EG&G Report (to be issued).

B. Test Results "Development of Dynamic in Situ Test Procedure for APSR Drive Mechanisms", 007 007 088 EG&G Report (to be issued).

C. H. Toffer "Criticality Control and Long Term Storage of Spent Fuel" IAEA-CN36/33, May 1977.

D. R. G. Nisle et al., "Fission-Product Build-up and Long-term Reactivity Effects" A/CONF 28/P/269, May 1964.

E. B. Lewis and V. Von Elbe "Combustion Flames and Explosions of Gases" Academic Press, 1951, p. 754.

4.0 EFFECTS ON SAFETY:

4.1 Documents which Define Safety of the System.

4.1.1 Sections of the SAR or its Amendments: Not Applicable.

4.1.2 Technical Specification Basis:

A. Technical Specification Change Request No. 37.

B. TMI Unit 2 Proposed Technical Specification Appendix A:

1. Section B3/4.1.1 Boron Control

2. Section B3/4.3.1 Neutron Monitoring Instrumentation

3. Section B3/4.6.1.1 Containment Integrity

4.1.3 Applicable NRC Regulatory Guides: Not Applicable.

4.1.4 Applicable NRC Standard Review Plan Section: Not Applicable.

4.1.5 Industry Codes and Standards: Not Applicable.

4.1.6 Previous safety evaluations:

A. Memorandum, C. Marotta (NRC/NMSS) to K. Kneil (NRC/NRR), "Recriticality Potential of TMI-2 Core", May 14, 1979.

B. "TMI-2 Criticality Evaluation Notebook" NPGD-TM-534 Babcock & Wilcox, December 1979. J. J. Woods, et al. *

C. E. W. Barr, et al., "TMI-2 Post-Accident Criticali y Analyses" TDR-049 GPU Service, August 31, 1979.

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

E. "Criticality Evaluation for PreHead Lift TER" 51-1131833-00 Babcock & Wilcox, March 4, 1982. *

F. "TMI-2 PreHead Lift Criticality Analysis" 32-1128350-03 Babcock & Wilcox, March 1982. *

G. R. D. DiSalvo, et al., "A Further Evaluation of the Risk of Recriticality at TMI-2", Office of Nuclear Regulatory Research, USNRC, April 4, 1980.

H. "Criticality Evaluation for PreHead Lift TER" 32-1128350-00 Babcock & Wilcox, March 1982. *

I. C. J. Mueller, et al., "An Evaluation of the Potential for and Consequences of Recriticality During Cleanup and Defueling at TMI-2" ANL/NRC-RAS 81-1, February 1991.

* Contains proprietary information.

4.1.7 Previous Technical Specification Change Requests:

A. Technical Specification Change Request No. 37.

4.2 Safety Function of Components Affected:

At the current time the excess reactivity of the reactor is controlled by the boron in the primary coolant. The calculations used to determine the boron concentration required to assure shut-down (Section 4.1.6) did not take credit for the presence of APSR poison in the core region. Hence, the systems affected by this test, the APSRs and their mechanisms, do not have a safety function. However, the test does have safety implications and they are discussed in Section 4.4 below.

4.3 Effect of Test on Safety Functions of Components Affected: Not Applicable. See 4.2.

4.4 Effect of Test on Probability of Occurrences or Consequences of an Accident:

4.4.1 Summary

This test will not affect the probability of occurrence or the consequences of an accident. Past safety evaluations of TMI Core 2 shutdown have not taken credit for the presence of the APSRs in determining shutdown. Hence, the reactivity changes associated with movements of the APSR poisons and the manner in which these changes are made, will not affect the conclusions of these past safety studies. Previous studies and studies reported herein, show that the shutdown margin provided by the current boron concentration is adequate to assure reactor shutdown even in the unlikely event core geometry is substantially changed as a result of APSR assembly motion causing fuel breakage and fuel redistribution.

No changes will be made to the primary coolant system or to supporting secondary or auxiliary systems (other than to provide power to the mechanism) for this test. Hence, there are no new or unreviewed safety questions with respect to boron control, effluent control, primary system integrity, containment integrity, etc. Fire monitoring provisions are provided in the test procedure.

4.4.2 Reactivity Shutdown Evaluation

The purpose of this discussion is to show that adequate reactor shutdown is assured for this test by the current primary coolant boron concentration of greater than 3500 ppm. This discussion will be separated into two sections. The first section will address the effect of this test on shutdown of the damaged reactor. The second section will address the test effects on fuel that may have been transported out of the core into other regions of the primary coolant system.

4.4.2.1 Reactor Shutdown

4.4.2.1.1 Previous Evaluations

Following the accident, calculations were performed by several organizations to assess the shutdown of the reactor. Some of these calculations are reported in references listed in Section 4.1.6 of this report. Specifically references A, B, C, D and G. These independent studies all support the contention that the TMI-2 reactor will remain shutdown at ambient temperature at boron concentrations of 3500 ppm.

The above calculations assumed various core damage models. In all cases the APSR assemblies were neglected. In most cases the other control rod assemblies and fixed burnable poisons were neglected. Extensive fuel rearrangement was assumed and parameter studies were performed to determine worst case conditions which were then used for the calculations.

The APSR testing will not invalidate the results of the above studies. As mentioned above, the shutdown provided by the APSR rods was not included in these studies. Further, the maximum reactivity addition that could result from the APSR motion is small compared to the shutdown calculated in the studies referenced above.

Additional TMI-2 core criticality studies have recently been performed at B&W and ORNL (J. Thomas). These studies investigated in further detail the shutdown of the reactor. The results of these studies have not yet been published, however, they support the conclusion of the above references that the reactor is shutdown at boron concentrations of 3500 ppm. The B&W studies investigated the reactivity effects of fuel enrichment and loading, fuel and fuel fine distribution, fission product decay, reduced temperatures, core structural materials, control rod worth (50% fuel damage model), changed VFs (Volume Fractions), burnup, etc. The ORNL study investigated the reactivity effects of fuel fine distribution within fuel rod lattices.

4.4.2.1.2 APSR Reactivity Worth

The TMI-2 Physics Test Manual (3.3.3.E) in Figure 2.2-19 presents values for the percent of total worth of the APSRs as a function of position. From this curve it can be seen that in an undamaged core, fully inserting these assemblies from their current position, will reduce their worth by 50%. Fuel redistribution in the mid region of the core and a less damaged lower core region would selectively reduce the worth of any partially withdrawn assemblies (4.1.6.D) and hence, reduce the absolute reactivity change resulting from their insertion or further withdrawal.

Figure 2.2-3 in the Physics Test Manual gives a maximum worth of $-.24\%\Delta\rho$ for the APSRs at O EFPD, 300°F, 1506 ppm boron. Table 2.2-2 indicates that the worth would increase slightly due to depletion (1.1). Using a total maximum worth of $.264\%\Delta\rho$ (.24%x1.1) and a 50% change in worth due to insertion, a value of $.132\%\Delta\rho$ is calculated for the change in reactivity resulting from the insertion of the APSRs. Higher boron concentrations will reduce the worth of the APSR's. From Figures 2.2-9 and 2.2-15, it can be seen that lower temperatures would also reduce the APSR worth.

The above calculations are for an undamaged core. If it is assumed that the upper half of the core is damaged, the worth of the APSR's could be less in the withdrawn position and hence, the reactivity increase on insertion would be less than calculated. In any case, even if an uncertainty as large as 300% is assumed for APSR worth due to core damage, the total reactivity insertion remains less than $.5\%\Delta\rho$.

4.4.2.1.3 In-Core Fuel Displacement

In addition to positive reactivity insertions that could result from APSR motion, it is conceivable that their motion could result in further fuel rearrangement. However, as the above referenced calculations did not address specific core configurations but rather, worst case studies, changes to the in-core fuel distribution resulting from APSR motion will be covered by these existing studies which confirm reactor shutdown at boron concentrations of 3500 ppm.

4.4.2.2 Out-of-Core Shutdown

4.4.2.2.1 Introduction

In evaluating the possible reactivity consequences of APSR motion the shutdown of out-of-core fuel must be considered. Out-of-core fuel transport mechanisms during APSR motion will be limited to short periods of natural circulation flow, coolant make-up flow, and gravity. Of these only gravity has the capability of moving any significant quantities of fuel. Hence, the only out-of-core region that could credibly see a change of fuel concentration as a result of APSR motion is the pressure vessel lower plenum.

It is unlikely that APSR motion will have any effect on out-of-core reactivity shutdown margins. Following the accident, fuel material that could have been swept out of the core would have been during the first day of the accident as a result of the initial reactor coolant pump switching and reflood transients. If any fuel is dislodged during the testing it most likely will remain in the core region. The bottom undamaged portion of the core, the core support structure, and the damaged fuel will act as a screen minimizing the possibility of fuel dropping into the pressure vessel's lower plenum.

4.4.2.2.2 Other Evaluations of Out-of-Core Shutdown

B&W in 4.1.6.B evaluated the possibility of critical fuel configurations in the lower plenum volume. These calculations have recently been supplemented by additional B&W studies not yet published. These new calculations included higher enrichments, more reactive fuel configurations, and other particle geometry. The results of these new studies show that for the 50% damaged core model, subcriticality is achieved at 3500 ppm boron for the maximum 2.98% fuel loading. In this case the B&W maximum damage model (top half of core damaged) would release the fuel from 30-2.98% assemblies. This calculation used a worst case VF and a hemisphere geometry (reference 4.1.6.E). Criticality could not be achieved with average enrichment fuel at 3500 ppm boron. Hence, for criticality to occur in this region, the maximum enrichment fuel must somehow segregate from the other enrichments.

4.4.2.2.3 Lower Plenum Shutdown Evaluation

The method used in this evaluation to show adequate out-of-core shutdown margin for APSR testing differs from past studies of TMI-2 in that it does not rely on calculations of a specific assumed geometry. Rather it uses operational data to determine a shutdown margin and compares this margin to reactivity changes that could result from fuel redistribution. This method evaluates the increased shutdown resulting from the addition of 500 ppm boron. This shutdown margin is compared to the amount of fuel that has to be added to result in criticality. This amount of fuel in turn is dependent on the amount of fuel initially in the lower plenum. Various initial concentrations of fuel are assumed and the fuel increments required for criticality are calculated.

It should be noted that both this method and past studies conclude shutdown margin exists. The results of this study show fuel transfers of the same order of magnitude (minimum, 66%), or larger, than those that are assumed to have already occurred, are required to cause criticality. Such transfers as a result of APSR testing are not credible.

4.4.2.2.3.1 Boron Concentration

On April 27, 1979, forced circulation of the primary coolant was terminated. At that time the boron concentration was about 2900 ppm (the chemistry logs show boron concentrations of 2869 ppm on April 25, 1979 and 2960 ppm on May 2, 1979). Because this concentration was established before forced circulation was terminated, this value can be considered representative of the entire primary coolant system. The current boron concentration in the primary coolant system is greater than 3500 ppm boron (the chemistry logs show a value of 3753 ppm on April 12, 1982). The chemistry logs show that the boron concentration has been greater than 3500 ppm since October 1979. The plant make-up rate has been such that many system volumes of primary coolant have been charged into the plant during the last two and a half years while natural circulation flow existed. Hence, it is considered that the current in-core boron concentration corresponds to the current chemistry sample results. Allowing for measurement accuracy (100 ppm) the boron concentration in the lower plenum is over 500 ppm boron greater than it was at the time forced circulation stopped.

4.4.3.2.3.2 Initial Shutdown

For the purpose of this evaluation it is assumed that the k_{eff} of any fuel in the lower plenum on April 27, 1979, was less than 1. This was at a primary coolant temperature of less than 180°F. The conclusion that the reactor was shutdown is supported by the following:

- Calculations referenced in Section 4.1.6 conclude that critical fuel configurations in out-of-core regions are unlikely. In addition to the analytical supposition that criticality in out-of-core regions is not credible, a mechanistic evaluation also shows that fuel transport sufficient to support criticality in out-of-core regions is improbable. Both the upper and lower core end fittings of the fuel assemblies provide a grillage that would preclude fuel transport of large fuel particles out-of-core. In addition surveillance activities to date have not located out-of-core regions containing the tons of fuel fines required for criticality.

- References 4.1.6.G and 4.1.6.I both conclude in separate studies that sustained criticality in the core region is not credible. These reports predict that in the event of local criticality, fuel dispersal would cause shutdown. In addition, they conclude, primary coolant temperatures and primary coolant activities would increase due to the energy required to establish core shutdown conditions. These increases would have been detectable. Such increases were not observed. These studies, although performed for in-core regions, would also apply to sustained criticality in the lower plenum.

- If the fuel in the lower plenum region was not subcritical at 3000 ppm boron, but instead was in a sustained critical configuration, it was not evident and did not present a safety problem. Primary coolant activity did not increase during the period of time when the plant was at or less than 3000 ppm boron and under natural circulation flow conditions. No increase was observed of the readings on the plant nuclear instruments. If sustained criticality was not evident and did not represent a problem at 3000 ppm boron, it will not present one at 3500 ppm boron. The increase of 500 ppm boron would not result in significant changes to the fuel reactivity coefficients and hence, essentially the same shutdown mechanisms will be available. As a result, the consequences of criticality would be expected to be the same, and not represent a safety problem. It is expected that coolant chemistry measurements would be the most sensitive means to detect sustained criticality in the lower plenum and an increase in boron concentration could be used to terminate the event.

4.4.2.2.3.3 500 ppm Boron Shutdown Reactivity Worth

The reactivity worth of the 500 ppm increase in primary coolant boron concentration is dependent on the core configuration. Its value will be lowest for a highly damaged fuel configuration hence, a highly damaged core geometry was used to assess its worth.

The boron worth is also sensitive to the VF (volume fraction) used in the calculations as well as assumed fuel enrichment. Marotta in reference 4.1.6.A, includes in his Figure 2, curves of k_∞ vs W/F (water fuel ratio) as a function of ppm boron for two enrichments, 2.96% and 2.31%.

This safety evaluation used Marotta's Figure 2 to determine the boron worth in the pressure vessel lower plenum. The W/F ratio giving the highest k_∞ at the highest boron concentration (3000 ppm) was used for this evaluation (0.8). In addition, an enrichment of 2.96% was used as this enrichment results in the most reactive configuration. A value of 2.75%$\Delta\rho$ for the worth of 500 ppm boron was obtained from 4.1.6.A's Figure 2.

4.4.2.2.3.4 Other Reactivity Changes Since April 27, 1979

In evaluating the net reactivity change of any fuel in the lower plenum since flow termination and boron concentration increase, it is necessary to consider the following:

- Xenon Decay:

  The TMI-2 Physics Test Manual (3.3.3.E), Figure 2.7-3, shows that xenon decay is completed in 80 hours. Hence, no reactivity change will result from xenon decay after April 27, 1979.

- Samarium Buildup:

  The TMI-2 Physics Test Manual (3.3.3.E), Figure 2.5-7, shows the buildup of samarium is completed 20 days after shutdown. Hence, no reactivity change will result from samarium buildup after April 27, 1979.

- Other Fission Product Decay:

  References 3.3.5.C and 3.3.5.D evaluate the effect of fission product decay on the reactivity of spent fuel. These studies conclude that the consequence of fission product decay is to increase shutdown. This increase in shutdown is conservatively not included in this evaluation.

- Temperature Change:

  The temperature defect for fuel in the lower plenum, from 180° to 70°F, is calculated to be .64%$\Delta\rho$. This value is derived from data contained in NPGD-TM-534 (4.1.6.B). This report states that for a VF of .63 and 2.6 w/o U-235 the temperature coefficient varies from $-.8 \times 10^{-4} \Delta\rho/°F$ at 2100 ppm to $-.5 \times 10^{-4} \Delta\rho/°F$ at 4000 ppm. Interpolation gives a value of $-.58 \times 10^{-4} \Delta\rho/°F$ at 3500 ppm. Table 5.12 in TDR-049 (4.1.6.C) shows that the magnitude of the negative temperature coefficient increases with higher VFs. Hence, B&W's use of a large VF is conservative. Temperature defect is primarily dependent on moderator and boron conditions. Fuel effects would be of lesser importance. Hence, this coefficient can be used for 2.96 w/o fuel. An additional conservatism in this calculation is the assumption that 180°F is the initial temperature. During the time boron concentrations were less than 3000 ppm and natural circulation flow existed, inlet temperatures were as low as, or lower than 155°F.

4.4.2.2.3.5 Net Shutdown Margin

In Section 4.4.2.2.3.3 above, it was shown that 500 ppm boron would provide an additional shutdown in out-of-core regions of 2.75% $\Delta\rho$. Section 4.4.2.2.3.4 above determined that the only significant source of reactivity addition since flow was terminated is the temperature change, conservatively identified as a decrease to 70°F, with a corresponding reactivity increase of .64% $\Delta\rho$. It can therefore be concluded that the net increase in shutdown, as a result of changes to plant conditions since flow was terminated, for any fuel that is located in the pressure vessel lower plenum amounts to 2.11% $\Delta\rho$.

4.4.2.2.3.6 Fuel Transfer Required to Offset Shutdown Margin

In Section 4.4.2.2.3.5 a minimum shutdown margin of 2.11% $\Delta\rho$ was identified for any fuel located in the lower plenum. In this section the amount of additional fuel required to offset this margin will be discussed. Calculations were performed by B&W to identify the necessary increase in fuel concentration required for a 2.1% increase in reactivity. These calculations are contained in reference 4.1.6.E. and are similar to those in reference 4.1.6.B. These results are determined from differences in reactivity and hence, are not as sensitive to errors in absolute reactivity. In addition because reactivity differences are used, many variables that effect absolute reactivity, such as pellet geometry, temperature, etc., are not significant in these calculations. The calculations were performed for fuel VFs of .55 (maximum reactivity) and 3500 ppm boron. It should be noted that the total calculated keff for the maximum enrichment fuel is greater than 1. Shutdown in these cases would result from geometry or poison effects not included in these calculations. These additional shutdown mechanisms would cancel out in the differential shutdown calculations. The average enrichment fuel cannot achieve criticality.

TABLE I

LOWER PLENUM SHUTDOWN EVALUATION
(3500 ppm boron)

Initial Number of Fuel Assemblies (Fuel Only) Assumed in Plenum	Additional Fuel Assemblies (Fuel Only) Required to Add 2.1% $\Delta\rho$	Geometry	Enrichment
7.5	*5	Sphere	Maximum (2.98%)
8.5	8.5	Sphere	Maximum
13.5	16.5	Sphere	Maximum
20	40	Sphere	Maximum
22	22	Hemisphere	Average (2.6%)
33.5	55	Hemisphere	Average
47.5	129.5	Hemisphere	Average

* Minimum condition for criticality, 12.5 total assemblies.

As can be seen from Table I, for criticality to occur as a result of APSR motion, large quantities of fuel must be transferred from the core to the lower plenum. These quantities are on the same order, or larger than, the quantities that had to be transferred during the accident. Such fuel transfer as a result of APSR motion is not credible. As stated earlier, the hydraulic forces, temperatures, etc., available during the accident to support fuel transport will not be available during APSR testing.

4.4.2.3 Conclusion

Based upon the above review, it is concluded that the existing boron concentration of greater than 3500 ppm in the primary coolant system, combined with existing operating procedures and systems, provides assurance that the reactor will remain shutdown during the APSR testing.

4.4.3 Fire Protection

4.4.3.1 General

For this test power will be applied to the control drive mechanisms. TV cameras will be positioned to monitor the area over the mechanisms to detect smoke or fire. The test procedure requires that these cameras be monitored during the test. This monitoring provides assurance that the existence of a fire would be detected in a prompt manner. Existing TMI-2 procedures would be adequate to assure that a fire would be promptly and safely extinguished.

4.4.3.2 Hydrogen

It is possible that the control drive housing and/or pressure vessel upper plenum may contain hydrogen gas. However, this gas, if present, would not constitute a hazard for this test. The composition in the gas spaces of the reactor coolant system (RCS) can be estimated from determinations of dissolved gas content of RCS water samples and the known solubility behavior of hydrogen, oxygen, and nitrogen in aqueous systems. If a gas phase is present in the RCS, then the partial pressure p_i of species i in the gas phase is related to the dissolved gas mole fraction x_i through the equation

$$p_i = K_i x_i \; ; \qquad (1)$$

where K_i is the Henry's law constant of the gas-water system at RCS temperature (38°C). Note that

$$P = \sum_i p_i + p_w \; ; \qquad (2)$$

that is, the total pressure is equal to the sum of the partial pressures of the gas components, plus the partial pressure of water under RCS conditions. (If a gas phase were present, P would be equal to the RCS pressure. Since the system pressure is considerably in ex ess of p_w, this term is hen eforth ignored.)

The mole frection y_i of component i in the gas phase under RCS conditions is given by

$$y_i = \frac{p_i}{P} = p_i / \sum_i p_i \; ; \qquad (3)$$

If Eq. (1) is substituted for the p_i, the result can be expressed in the form

$$y_i = K_i x_i \; / \; \sum_i K_i x_i \; ; \qquad (4)$$

By inserting the appropriate Henry's law constants at 38°C from "Handbook of Chemistry and Physics (43rd edition, 1961-1962, p. 1708), the mole fraction of oxygen in the gas mixture in the RCS that is comprised of oxygen, hydrogen, and nitrogen is given by

$$y_{O_2} = \frac{4.04 \; x_{O_2}}{4.04 \; x_{O_2} + 5.72 \; x_{H_2} + 7.51 \; x_{N_2}} \qquad (5)$$

or, since ratios of concentrations (cc/kg) are involved,

$$y_{O_2} = \frac{4.04 \; C_{O_2}}{4.04 \; C_{O_2} + 5.72 \; C_{H_2} + 7.51 \; C_{N_2}} \qquad (6)$$

where the C_i denote concentrations of the corresponding gases dissolved in the RCS water.

From "Combustion, Flames, and Explosions" (Section 3.3.5.E), mixtures of hydrogen, nitrogen, and oxygen are nonflammable at O_2 compositions of 5% or less. Hence, flammable mixtures within the RCS are not possible if

$$y_{O_2} \leq 0.05 \ , \qquad (7)$$

Substitution of Eq. (6) into this expression yields

$$C_{O_2} \leq 0.075 \ C_{H_2} + 0.098 \ C_{N_2} \ , \qquad (8)$$

For purposes of simplification, the dilution effect of hydrogen can be ignored. This leads to the conservative criterion

$$C_{O_2} \leq 0.098 \ C_{N_2} \ , \qquad (9)$$

that is, flammable concentrations in the RCS are avoided so long as the concentration of dissolved nitrogen is approximately a factor of ten greater than that of oxygen (in cc/kg).

During the period of APSR motion, primary coolant chemistry will be monitored every 24 hours. If the ratio of C_{O_2}/C_{N_2} becomes greater than 0.098, testing will be stopped temporarily until the ratio is decreased below 0.098. Since C_{N_2} is expected to range about 6 cc/kg, this corresponds to a dissolved oxygen concentration of 0.59 cc/kg (0.48 wppm). If C_{N_2} rises above the 6 cc/kg concentration level, the permissible dissolved oxygen concentration can be similarly increased.

4.5 Effect on Equipment Important to Safety: No equipment Important to Safety (ITS) is affected by this test (see Section 4.2).

4.6 Possibility for an Accident or Malfunction of a Different Type

Movement of the APSRs does not introduce a possibility of an accident or malfunction that has not been evaluated in previous safety reports.

4.7 Decreased Margin of Safety:

This test does not decrease the margin of safety as defined in the TMI Unit 2 Proposed Technical Specifications. Section 4.4 of this report shows that the existing shutdown provides adequate margins to accommodate any reactivity changes resulting from APSR motion.

4.8 Violation of Technical Specifications:

No Technical Specifications are violated. This test is covered by the action of Technical Specification Change Request No. 37.

4.9 Change of Licensing Requirements or Regulations:

No licensing requirement or regulation is exceeded as a result of this test. All accidents postulated are enveloped by prior safety studies and the Technical Specifications will not be violated.

4.10 Radiological Safety:

The radiological aspects of this test are limi ed to the radiation exposure received attaching the test instrumentation to the control rod mechanisms and moving the TV cameras. The actual test will be conducted from outside the containment building. No effluents will be generated as a result of this test. The standard GPU Nuclear radiological control procedures will be utilized for this test. It is estimated that an exposure of less than 5.0 man-rem will result from this test.

5.0 Conclusion:

It is proposed to directly energize each ASPR mechanism, one at a time, to withdraw the leadscrew a short distance and then fully insert it into the core. This will be done from the control rod drive mechanism cabinet in the cable room. Rod motion will be monitored by monitoring for pole slippage. This slippage will be detected by either acoustic or electrical measurements. Individual rod position indicators will also be used if they are available.

The purpose of this test is two fold:

A. The ability to move some or all of the APSRs will provide an insight into the extent of core and upper plenum damage. This early insight will provide time to factor this information into plans for subsequent inspections, head and upper plenum removal and core removal.

B. Prior to head removal it is necessary to decouple the APSR lead screws. This can be most easily accomplished if the rods are fully inserted. Hence, inserting the APSRs into the core is one step in the recovery process.

An evaluation of the effects of this test on the shutdown of the reactor and out-of-core fuel concentrations shows that:

A. The total reactivity effect of APSR motion is negligible compared to observed shutdown margins.

B. The extent of in-core and out-of-core fuel rearrangement required to override the observed shutdown margin is so large as not to be credible.

Based upon this evaluation it is concluded that:

A. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety is not increased as a result of this APSR testing.

B. APSR testing does not introduce the possibility of an accident or malfunction of a different type than previously evaluated for TMI Core 2. The safety considerations are primarily associated with reactivity shutdown. Studies reported in this evaluation show that the existing boron concentration provides adequate shutdown for these tests.

C. The margin of safety as defined for the basis of the Proposed Technical Specifications is not reduced. The existing boron concentration ensures that the core will remain subcritical under all possible conditions which may exist during this test.

The above evaluation was made for moving the APSRs. Depending on the results of this test, it may be desirable to extend the testing to selected safety/regulating rod assemblies. The above conclusions would also apply to testing these assemblies. Actual motion, one at a time out and back in, would be limited to small distances (less than one half inch) which would have the potential of in roducing less total reactivity than the APSR testing could introduce (3.3.3.E). B&W calculations estimate that the total worth of all safety/regulating rod assemblies in the damaged core is 3.3%$\Delta\rho$. The worth of the additional 500 ppm boron in the damaged core is such that criticality due to movement of a single safety/regulating rod assembly could not result in criticality even under accident conditions. Other test implications would be unchanged.