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A FURTHER EVALUATION OF THE RISK OF RECRITICALITY AT TMI-2

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ABSTRACT

This report reviews previous studies related to the probability and consequences of criticality at the damaged Three Mile Island Unit-2 reactor. More detailed assessments are performed to confirm the adequacy of those studies and to provide additional insight into ways to minimize risk from criticality. The most important conclusions of this study are:

1. The most probable mechanism for criticality, boron dilution, is a slow enough process that with appropriate instrumentation and procedures, the approach to criticality can be detected and corrected. To the extent that boron concentration in excess of 3500 ppm can be ensured, the probability of criticality is further minimized.

2. The most likely direct radiological consequence of criticality is increased dose rates inside containment. For the more realistic and more probable criticality events studied, off-site consequences are nonexistent. More conservative assumptions regarding the nature of the criticality, combined with multiple failures of engineered safety features are required before one calculates detectable health effects. Even then, the consequences, as expressed in terms of the probability of latent cancer fatality, appear to be very small compared to the observed incidence of cancer death. To the extent that core cooling and containment integrity can be maintained, the consequences of criticality can be further minimized.

CONTENTS

1. INTRODUCTION

On February 28, 1980, a special task force formed by NRC's Acting Executive Director for Operations reported its findings regarding the cleanup activities at Three Mile Island.[1] Among its recommendations were that the "staff reevaluate the potential for recriticality and ensure that adequate procedures and equipment are avail ble to prevent its occurrence."

On March 10, 1980, the Director, Probabilistic Analysis Staff of NRC's Office of Nuclear Regulatory Research directed[2] the authors to perform an independent assessment of the risk of criticality and to prepare this report. Specifically, we were to "review work already done on this matter by the Kemeny Commission staff, the Rogovin inquiry and NRR. Consider the mechanisms by which boron could be lost from the core region so that recriticality might occur. Evaluate the probability of criticality occurring, the rate at which criticality could be approached and the likely consequences of such an occurrence." Our findings and recommendations follow.

2. CURRENT STATUS OF CRITICALITY CONTROL AT TMI-2

As of March 31, 1980, the damaged core of TMI-2 is subcritical as verified by the single remaining excore source range neutron detector. Believed to be maintaining subcriticality is a boron concentration (as boric acid) of 3850 parts per million (ppm) in the Reactor Coolant System (RCS) water. This is measured weekly at a location upstream from the letdown coolers approximately 200 feet from the core. A technical specification lower limit of 3500 ppm boron has been established. There is essentially no flow of water in the RCS except during short periods associated with "burping" in the steam generators. The pressure and average temperature in the RCS are about 280 psi and 150^{0}F respectively.

The core is presumed to have been uncovered for up to two hours during the accident of March 28, 1979. A major portion of the Zircaloy was oxidized, and the fuel, control, and burnable poison rods experienced thermal transients beyond their design conditions. A cone of failed, oxidized fuel rods is believed to extend from the top of the core to eight feet downward.

The control rods (silver-indium-cadium alloy) entered the core seconds into the event. Their current status is uncertain, but melting of at least the top third should have occurred as a result of the thermal transient. Much of the control rod material may be retained in the outer and lower regions of the core. Some of the boron in the fixed burnable poison rods (B_4C-Al_2O_3) is probably lost since boron is known to leach out when exposed to water in a radiation environment.

3. SUMMARY OF ANALYSES TO DATE

3.1 Probability of Recriticality

Criticality analyses of the TMI-2 core have been made by the NRC staff (1,3,4,5), Babcock and Wilcock(6), Brookhaven National Laboratory (7), General Public Utilities (8,9), and Oak Ridge National Laboratory (10). The important models and results of these analyses are summarized in Appendix A.

A reevaluation of these analyses yields the following conclusions:

(i) - The keff of the core with 3500 ppm boron is conservatively estimated to be less than 0.90. With at least 3500 ppm boron, the core will remain subcritical in any physically reasonable rearrangement of the fuel even in the total absence of control rods or burnable poison.†*

† In a recent NRC memorandum (11), Marotta points out that the ORNL analysis(10) does not assume the most reactive core configuration given our current understanding of the core's physical condition. He recommends using a higher reference keff (0.944 at 3000 ppm boron) for boron dilution studies. Using 100 ppm as equivalent to - 1% $\Delta k/k$, the higher reference keff yields k=0.894 at the technical specification lower limit of 3500 ppm boron.

* Given the uncertainty regarding the status of control materials and burnable poisons, these analyses give no credit for their contribution to criticality control. As a result, the calculated concentrations of boron required to maintain subcriticality are overestimated, perhaps by as much as a factor of two.

(ii) - The calculational methods and nuclear data used in the analyses are adequate. The methods used by NRC/NMSS[4], B&W[6], and BNL [7] have been tuned to experimental data through the years. The ORNL analysis [10] includes calculations of the TMI-2 core at startup with all rods out, critical at zero power, and RCS conditions of 220 psi, 532°F and 1490 ppm boron. The ORNL calculation underpredicts criticality by about 1.5% $\Delta k/k$. This was taken into account when determining that 3500 ppm boron was adequate to prevent criticality.

(iii) - The potential for small unborated or underborated volumes of water to enter the core without the benefit of mixing must be considered as well as the more commonly addressed concern of a well-mixed gradual dilution. The affect of assuming zones of the core with lower boron concentrations was studied by Marotta.[4] Introducing 1000 ppm borated water into the outer regions of the core would result in criticality. Introducing a coherent mass of unborated water with a volume of 3 ft^3 into the core would also result in criticality. This latter calculation is supported by data from the Westinghouse Reactor Evaluation Center in 1967.

(iv) - The analyses make no quantitative estimates of the probability of achieving the conditions necessary for criticality. The major concern is the introduction of water with less than 3500 ppm into the core. The studies generally conclude that with appropriate precautions related to sampling and introducing water into the RCS, the approach to criticality is detectable and avoidable. Many recommendations designed to minimize the probability of criticality have been made with these thoughts in mind.

3.2 Consequences of Criticality

Thompson and Beckerly[12] have reviewed reactor accidents involving criticality or reactivity changes. A summary table from Reference 12 which includes total fissions and estimates of radiation dose is reproduced here as Table I. Except for NRX and SL-1, the events described resulted in little or no radiation dose. These data must, however, be viewed cautiously if one wishes to extrapolate them to TMI. Special consideration must be given to major differences in core design, in the initial configuration of the core, in the design and availability of engineered safety features and any other factors which are unique to TMI in its current configuration.

Another key reference in regard to accidental criticality is the well known study by Stratton.[13] The TMI-2 core is in the category of inhomogeneous water-moderated cores reviewed by Stratton. Two types of accidental criticality are reviewed. Accidents caused by the sudden insertion of reactivity (such as Borax 1, the Spert tests, and SL-1) apppear to be limited by the rapid, almost adiabatic production of heat in the core. The power curve looks like a sharp peak. Typically 5×10^{18} fissions occur, corresponding to a production of 158 MJ. For a large core, this might be an order of magnitude larger.

Some accidents have involved slow approaches to criticality in which the reactor does not go prompt critical. One such example is the NRX accident of December 12, 1952. After the reactor attained criticality it would rise in power until either of two conditions was met: (i) the reactor became unstable and eventually overheated through loss of cooling; or (ii) the available reactivity was used up and the reactor operated at a steady power.

An inexorable increase in reactivity through continued removal or boron would probably lead to unstable boiling since all but the more optimistic evaluations of reactivity indicate considerable potential for added insertion. It is hardly conceivable that such an increase would occur except in the absence of all precautions plus deliberate dilution of the cooling water. Nevertheless, the advantage of early warning of reactivity increase from neutron detectors as well as effective monitoring of boron concentration is that, even if dilution does occur, the reactivity increase can be stopped and reversed before unstable core performance leads to more fuel melting.

TABLE I. REACTOR ACCIDENTS INVOLVING CRITICALITY OR REACTIVITY CHANGES (FROM REFERENCE 12)

Date	Location, name	Active Fuel Coolant, Moderator	Geometry	Total Fissions	Cause (C) Quenching Mechanism (Q)	Person (P) Radiation dose (R)	Damage	Ref. in this chapter
Dec. 1949	LASL (water boiler) crit. 1944, 1950	1 kg U^{235}, $UO_2(NO_3)_2$ in 13.6 liter H_2O	Sphere, graphite reflector	$3\text{-}4 \times 10^{16}$	(C) Manual withdrawal of 2 control rods (Q) Expansion and rise of neutron temperature	P-1: R-2.5r gamma	None	Sec 3.2
12 Dec. 1952	Chalk River, Canada NRX, criticality 1947, full power May 1948	Natural uranium rods H_2O-cooled, D_2O-moderated	Rod lattice, graphite reflected	0.6×10^{20}	(C) Control rod mal-operation, safety circuit failure-complex (Q) Dump of D_2O moderator	P-none, except in clean-up; many P got small doses, highest 17r, most less than 3.9r	Core badly damaged, removed, replaced	Sec. 3.3
22 July 1954	NRTS Idaho BORAX I (transient tests 1954)	93% enriched U^{235} in U-Al plates (MTR type), H_2O-moderated	H_2O reflector, swimming pool excursion reactor	4.68×10^{18}	(C) Estimate of expected excursion low (Q) Steam void disassembly	P-none R-none	Core destroyed	Sec. 3.4
3 Oct. 1954	Hanford Production Reactor First one critical September 1944	Natural uranium rods H_2O-cooled, graphite-moderated	Process tube type—large graphite reactor	Local over-heating	(C) Water leak changed reactor pattern, short period occurred (Q) Control rod changes then scram	P-none R-none	Some fuel elements failed or were damaged	Sec. 3.5
4 Jan. 1955	Hanford — KW Reactor	. . . . same	. . . same . . .	Fuel failure, local melting	(C) Blockage of cooling water in process tube — initial start-up power decrease noted — rods withdrawn	P-none R-none (Q) Scram on over-pressure	Graphite channel removed by hole cut in shield	Sec. 3.5
Jan. 1955	Hanford Production Reactor	. . . . same	. . . same . . .	No over-power	(C) Misestimate of ρ-instruments (Q) Rod run in by operator	P-none R-none	None	Sec. 3.5
29 Nov. 1955	NRTS Idaho EBR-I Mark-II Operations in 1951	0.5 in. U^{235} rods, NaK-cooled fast reactor	Compact core Nat. U blanket	4.7×10^{17}	(C) Estimate of expected results low — earliest scram attempts not effective (Q) Shutoff by second scram; fuel bowing a factor	P-none R-minor	Core melted, little [illegible]er contamination	Sec. 3.6
9 Oct. 1957	England-Windscale Operations in July 1950	Natural uranium rods air-cooled, graphite-moderated	8-sided stack of graphite 50 x 50 x 25 ft 25 ft fuel channels	Graphite-uranium fire	(C) Wigner energy release, U-burning triggered by nuclear overheating (Q) Flooding with H_2O	P-none serious R-widespread radioactivity, milk over 200 mi^2 area destroyed	Severe core damage reactor not rebuilt	Sec. 3.7
18 Nov. 1958	NRTS — HTRE-3 crit. October 1958	Enriched uranium gas-cooled, solid moderator	Horizontal cylinder	Not known	(C) System on auto-control with faulty instrumentation (Q) Meltdown slump and/or scram	P-none R-some site contamination	Core melted, basic system undamaged	Sec. 3.8
24 July 1959	Santa Susanna, Cal. SRE crit. 1957 at power May 1958	2.8% U^{238} slugs in SS-clad rods Na-cooled graphite-moderated	Pseudo-cylinder, graphite reflector	2×10^{19} (in last minute)	(C) Coolant channel blockage by impurities overheating, perhaps fuel bowing (Q) Manual scram	P-none R-release was ~0.3% of core activity inventory	12 of 43 elements melted, core removed and replaced	Sec. 3.9
3 April 1960	Waltz Mill, Pa. WTR crit. 1959	93% enriched U^{235} U-Al plates-cylinder H_2O-cooled-moderated	Pseudo-cylinder, water reflected	Overheat of one element	(C) Undercooled, perhaps faulty fuel — negative auto control response (Q) Manual scram	P-none R-minor release to site	1 element melted, $\$10^6$ to clean up damage	Sec. 3.10
3 Jan. 1961	NRTS SL—1 crit. August 1958	93% enriched U^{235} Al-U plates, boiling, H_2O-cooled-moderated	Pseudo-cylinder 5 rods, B-Al strips control	1.5×10^{18}	(C) Manual withdrawal of central control rod (Q) Expansion, boiling, core evaporation	3P-all fatal R > 800r/hr in bldg.; in recovery 14P got R > 5r	Core destroyed, vessel rose 9 ft, reactor dismantled	Sec. 3.11
5 Nov. 1962	NRTS SPERT I (destructive tests)	93% enriched U^{235} Al-U plate type H_2O-moderated	Pseudo-cylinder with transient ρ-rod, open tank		(C) Planned experiment—reactivity transient as planned, energy release effects more destructive than planned (Q) Expansion, steam void, no melting involved	P-none R-minor site contamination	Core destroyed (destruction planned on this test or next)	Sec. 3.12

A best estimate of the stable power that can be reached at or near atmospheric pressure can be made by extrapolation from the past natural circulation boiling water experiments, such as EBWR. A 20% average void in the core corresponds to about 100% quality in the hot channel exit. A steam velocity of about 0.3 m/s is commonly observed, and we can conservatively estimate a mean bubble rise of 2 m. An energy balance then yields an estimate of 2/3 Mw, or since this is an order of magnitude estimate, about 1 Mw. This is in accord with the experience cited by Thompson and Beckerly[12] and by Stratton.[13]

At higher system pressure, higher powers can be attained. As a best estimate, assuming two-thirds of the control rod material is effective, there might be 5% excess reactivity if all boron were removed. Assuming 2% of Δk for Doppler and temperature defect, this would yield about 12% average void at high pressure (2200 psi) or on the order of 100 MW. At the current TMI-2 system pressure (280 psi), the power level would be about 15 MW. This is quite approximate; each 1% Δk beyond the 2% Δk to reach temperature represents about 30 MW.

Appendix 3 in NRC's Task Force Report "Evaluation of the Cleanup Activities at Three Mile Island"[1] attempted to bound the radiological consequences of a recriticality event by comparing it to the WASH-1400 sequence TKQ. In this sequence, a transient occurs while the reactor is critical, followed by failure of the reactor protection system and by failure of the subsequently opened relief valve to close. This results in core melt. Containment engineered safety features operate to remove heat and radioactivity from the containment atmosphere. The fission product inventory assumed for these calculations is the current one at TMI-2.

The results are reproduced here as Figure 1. It shows the probability per year of a person at a given distance from the reactor site suffering a latent cancer fatality assuming the event, in this case the TKQ sequence, has occurred.* TKQ is presumed to be bounded by the curve labeled "CASE 3 TMI-2 + 1 YR" if the containment is unisolated. If isolation is accomplished, the curve labeled "CASE 2 TMI-2" is more representative of the consequences. In either case, the probability of latent fatality to people more than five miles from the site appears negligible compared to more common causes of accidental death. For individuals at the site, the probability of latent fatality is one to two orders of magnitude higher. The authors of Reference 1 believed the statistical uncertainty in the predictions of nuclear accident risk in Figure 1 to be no more than a factor of 100.

For the sake of later comparisons, we have modified Figure 1 to include the normal incidence of cancer fatality and the mortality rate from all causes of death.

4. OUTSTANDING QUESTIONS AND FURTHER EVALUATIONS

This section focuses on unresolved standing questions or newly identified questions related to risk from criticality and contains additional analyses we performed relevant to their resolution.

4.1 Probability of Criticality

Though boron dilution is viewed as the most probable cause of criticality, there are other ways in which soluble boron might be lost from the core region. Figure 2 is a simplified logic tree indicating mechanisms by which such losses might occur. We made no attempt to quantify this tree, i.e., to evaluate the quantitative probability of criticality occurring. Rather, it

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* The radiological source terms were not large enough to result in any acute fatalities. No estimates of land contamination or psychological effects were attempted.

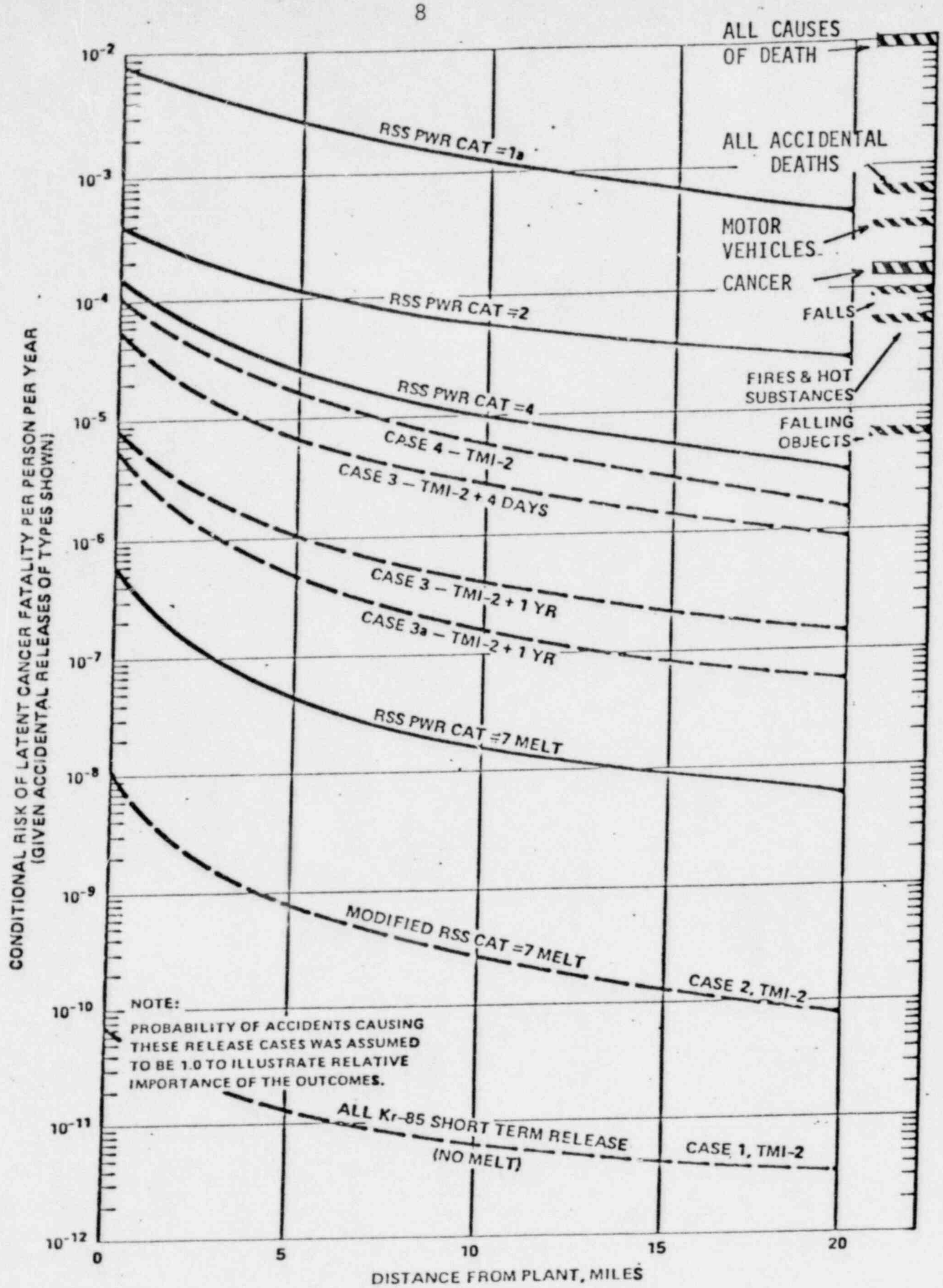
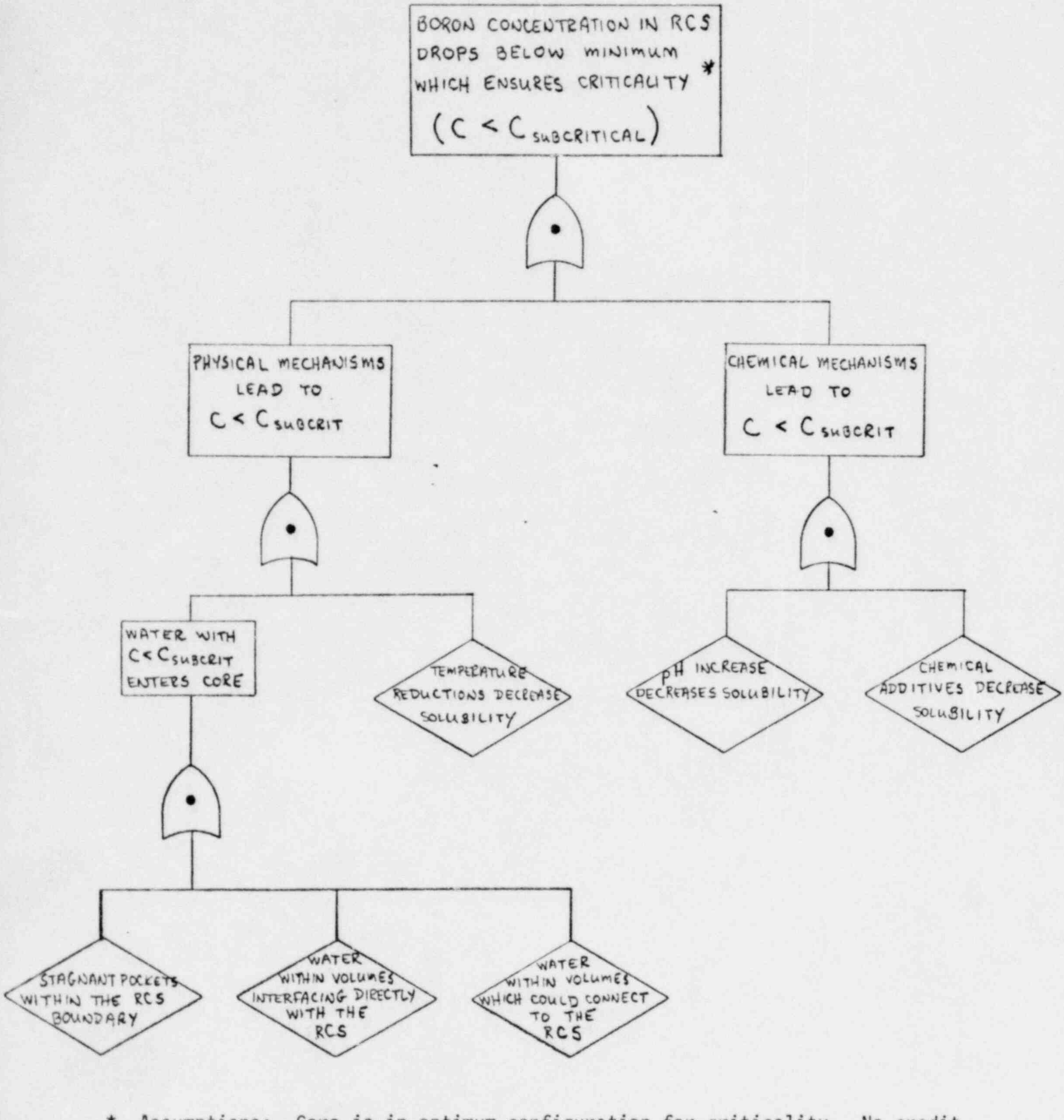


Figure 1. Conditional risk of latent cancer fatality as a function of distance for a spectrum of accident sequences. (After Reference 1, Appendix 3, Figure 3-3).

Figure 2. Simplified logic tree indicating mechanisms by which boron could be lost from the core region so that recriticality might occur.



* Assumptions: Core is in optimum configuration for criticality. No credit for control rods or burnable poison. $C_{subcritical}$ is the technical specification lower limit of 3500 ppm.

was believed that by taking proper precautions, the conditions necessary and sufficient for recriticality to occur could be precluded. Evaluations were performed and recommendations developed to maximize assurance that this was the case.

The mechanisms of Figure 2 fall into four areas which are discussed further below:

- concentration effects
- temperature effects
- pH effects
- chemical reactions.

4.1.1 Concentration Effects

There are at least three potential sources of water with lower than desirable concentrations of boron which might enter the core. There are stagnant pockets within the current RCS boundary; water volumes interfacing directly with but isolated from the RCS boundary; and water volumes which could enter the RCS through suitable connections.

Examples of stagnant pockets could be letdown lines, the pressurizer, portions of the RCS drain system and other regions which are outside the natural circulation flow path. There is no way to measure the boron concentration in these locations, though it is presumed that the entire RCS has the same boron concentration as that measured near the letdown coolers. Since these stagnant regions were originally borated, since they represent a small fraction of the RCS inventory, and since they would have the opportunity to mix with the RCS inventory before entering the core, there is no reason to suspect that they pose a problem.

An example of a water volume interfacing directly with, but isolated from the current RCS boundary is the pipe run in the low pressure injection system between the check valve nearest the reactor vessel and the motor operated isolation valve outside containment. The volume here is substantial (approximately 950 gallons of water), and inducing flow in the line would deliver this water to the downcomer and into the core region with little opportunity for mixing. This water is normally borated (typically 1500-2200 ppm) since it is part of the low pressure injection system.

There are many examples of water volumes which might be potentially aligned for one reason or another to deliver into the RCS. Examples are the refueling water storage tank, the containment sump water and less obvious sources such as fire hoses.

Of particular interest at this writing is a mini decay heat removal system (MDHRS) having a design capacity of 200 kw which is scheduled to be put into service in mid-April, 1980. Its design pressure is 235 psi and it has been hydro-tested to 350 psi. When operating, this system will induce a flow of 150 gallons per minute in the primary system. The MDHRS will tap into the existing residual heat removal system at the motor operated isolation valve outside containment. The flow will pass through the two check valves in the low pressure injection line and enter the RCS near the downcomer. The system will receive flow from the RHR outlet in a hot leg. It will contain a water sampling port approximately 50 feet from the core. Plans are being developed for monitoring the boron concentration from this location.

The MDHRS is a closed cooling loop containing approximately 200 gallons. If it were assumed that water in the MDHRS containing no boron and water in the pipe run to the RCS containing 1500 ppm boron were added to and mixed with the 30,000 gallons of water in the pressure vessel, the boron concentration would decrease from 3850 ppm to 3750 ppm, still well above the technical specification lower limit. (See Appendix B for a more detailed analysis).

4.1.2 Temperature effects

The solubility of boric acid in water decreases as water temperature decreases.[14] At the current RCS water temperature of 150^{0} the solubility limit is 24000 ppm. Temperature inside the containment is typically 75^{0} F, and water in thermal equilibrium there can sustain 8900 ppm in solution. At 32^{0} the solubility limit is 4400 ppm. Therefore, decreases in soluble boron concentration resulting from temperature decreases do not appear to pose a problem.

4.1.3 pH effects

Boric acid (H_3BO_3) is a weak acid in water. The solubility of boric acid in water is affected by the hydrogen ion concentration. Additions of base, such as NaOH, to the RCS water would increase the solubility of boron. Additions of strong acids, such as nitric acid (HNO_3) would decrease the boron solubility. However, large amounts of strong acid would have to be added before significant decreases in soluble boron concentration were observed. At this time there are no foreseeable circumstances under which such additions would occur.

4.1.4 Chemical reactions

Borate compounds are among the most soluble of all salts. Exceptions are the borate salts formed by the alkali metals calcium and magnesium. Large additions of aqueous solutions of these cations could precipitate boron out of solution. At this time, there are no foreseeable circumstances under which such additions would occur. However, to be prudent, any chemical additives contemplated for introduction into the RCS with the core in place should be tested for their compatibility with soluble boron.

4.1.5 Approach to criticality

The rate at which criticality is approached is determined primarily by the rate of decrease of boron in solution. Of the mechanisms described above, the concentration effects, i.e., boron dilution, appear to dominate the probability that criticality will occur.

Excore source monitoring provides a direct measure of the approach to criticality. At this time only one such instrument is available and its ability to continue functioning in the severe environment it has endured is uncertain. It is prudent to restore the neutron monitoring capability close to the core, but this requires access to the reactor head area. An interim measure might be to monitor the radioactivity level of the reactor coolant as it circulates through the MDHRS once that system is in service.

An alternative but indirect measure of the approach to criticality is the boron concentration. In order for this parameter to be a valid measure, one would have to be assured that the actual concentration of boron in the core region is accurately represented by the concentration measured at the sampling port.

If a pocket of relatively unborated water were forced through the core by some unspecified mechanism, the approach to criticality could be too quick for the operator to detect and prevent. The likely result of this, however, would be a local criticality of short duration. As will be shown in Section 4.2, such an event is relatively inconsequential in terms of its radiological impact.

By virtue of the large volume (90,000 gal) of the RCS, the current high boron concentration, and the likely low flow rates at which water would be circulated, it would require from days to months to decrease the boron concentration of the entire RCS to below critical limits.(1) This should allow ample time for the operators to recognize and prevent the approach to criticality. The probability that boron dilution is detected prior to criticality increases with boron sampling frequency.

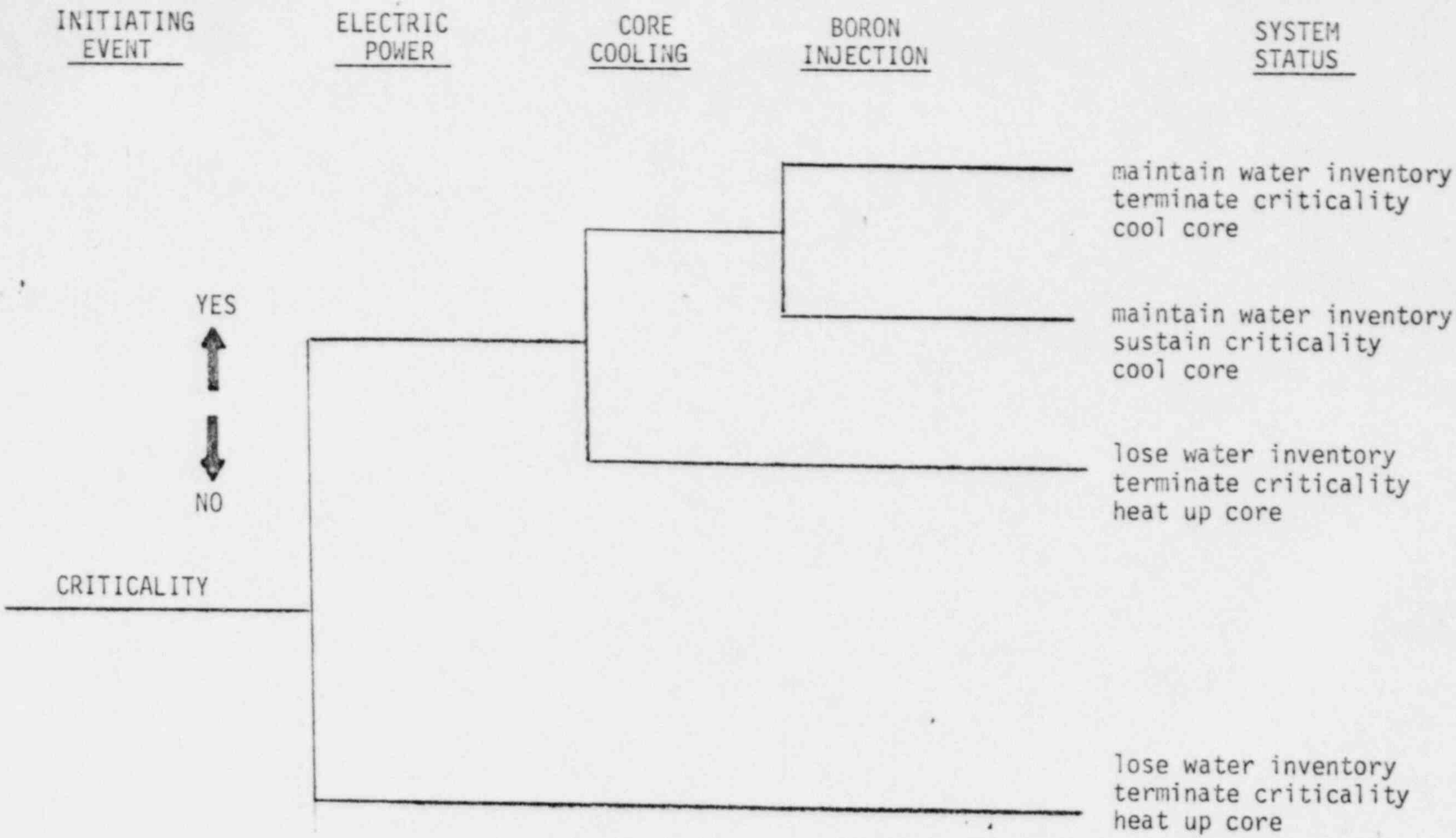
4.2 Consequences of Criticality

Though all practical measures should be taken to prevent criticality, it is assumed here that sufficient boron is lost from the core so that criticality occurs. Figure 3 is a simplified event tree which portrays a spectrum of possible outcomes shaped by the availability of key safety systems. The tree is not quantified because of insufficient data on the availability of these systems under the peculiar circumstances at TMI, though it is believed they would be operable more often than not. Furthermore, the status of the plant, the nature of the criticality and the radiological consequences are interdependent and vary with time. Nevertheless, the tree is useful in providing a framework for subsequent analyses and in making some qualitative judgements regarding relative probabilities and consequences of events.

In this analysis consequences are expressed in terms of the energy and fission products generated during criticality and in terms of the potential effects which the former has on dispersing the latter. We made no new calculations of radioactivity dispersion in the environment or subsequent health effects. Rather, where appropriate, estimates were made relative to those of Figure 1.

Two modes of criticality are considered: transient and sustained. A transient criticality (or pulse) might be induced by a slug of cold unborated water being pushed through the core by a column of borated water. Such an event might occur, for example, when putting a new system into operation. Table II indicates the character of the transient criticality assumed in this analysis in terms of the power achieved, the fraction of the core involved and the duration of the transient.

Figure 3. SIMPLIFIED EVENT TREE TO ASSIST IN DETERMINING CONSEQUENCES OF CRITICALITY



Sustained criticality is the other mode of criticality considered. This might be induced by a continuous flow through the Reactor Coolant System (RCS) of water having a low or zero concentration of boron. Once the boron concentration decreased to about 1500 ppm, the core is assumed to go critical. Further reductions in boron concentration would increase the ultimate power level achieved. Criticality would be detected by RCS core neutron monitoring, pressure, temperature and radiation detectors. Given zero or low forced flow or inability to restore the boron concentration, pool boiling would occur in the RCS. Some heat would be transferred via natural circulation to surrounding structures and to secondary heat sinks. Energy would be stored in the RCS water until the availability of a pressure relief path from the RCS allowed energy to be transferred via vaporization of water. An equilibrium power level would be reached whose magnitude would depend primarily on boron concentration, fuel temperature and voiding in the core. (See Section 3.2). Without makeup flow, the water level would decrease and this loss of moderator would eventually terminate criticality. Of course, restoration of boron could also be used to terminate criticality. The energy stored in the fuel and the fission product decay heat balanced against available heat removal mechanisms would determine the driving force for heating the fuel and for dispersing radioactivity during and subsequent to criticality.

The approach to criticality and the course of subsequent events depend most upon those factors, including operator actions, which affect the time dependent concentration of boron in the core region. For the purpose of providing quantitative indicators of consequences, three cases of sustained criticality were assumed as described in Table II. The power level and time at power are the variants. The practical basis for the assumed cases is that boron dilution goes undetected long enough to reach criticality. Criticality is detected within minutes after it occurs and operator action halts further dilution. The RCS and containment are successfully isolated. It is assumed, however, that efforts to

TABLE II. ENERGY ACCOUNTING FOR CRITICALITY EVENTS

	TRANSIENT (PULSE) CRITICALITY	SUSTAINED CRITICALITY		
		CASE A	CASE B	CASE C
Energy Generation Rate (MW)	2772	27.7	277	277
Fraction of Core Which is Critical	0.1	1.0	1.0	1.0
Time at Power (MIN)	1	60	60	600
Total Energy Generated (MJ)	1.7×10^{4}	1×10^{5}	1×10^{6}	1×10^{7}
Energy Dissipated (MJ)				
Heat Fuel to Maximum Power Level	5.1×10^{3}	9.8×10^{3}	1.7×10^{4}	1.7×10^{4}
Heat RCS Water to RHR Pressure Relief Conditions	8.0×10^{4}	8.0×10^{4}	8.0×10^{4}	8.0×10^{4}
Heat RCS Water to RCS Pressure Relief Conditions	N.A.	3.9×10^{5}	3.9×10^{5}	3.9×10^{5}
Heat RPV to RCS Pressure Relief Conditions	N.A.	N.A.	5.2×10^{4}	5.2×10^{4}
Vaporize Half of RCS Water Inventory	N.A.	N.A.	4.7×10^{5}	4.7×10^{5}
(SUBTOTAL)	N.A.	N.A.	1.0×10^{6}	1.0×10^{6}
Energy from Criticality Remaining to be Dissipated to Prevent Fuel Melting (MJ)	0	0	0	9.0×10^{6}
Energy to Heat Fuel to Melting from Equilibrium Power Level (MJ)	N.A.	N.A.	5.4×10^{4}	5.4×10^{4}
Decay Heat Power at Shutdown (KW) (Including 164KW Prior to Recriticality)	166	174	264	1164
Time to Fuel Melt at Decay Heat Power (DAY)	N.A.	N.A.	4.8	1.0

increase the boron concentration are nonproductive for the specified time. Sustained criticality Case A is our realistic estimate of equilibrium power level at the current system pressure of 280 psi. Case B is our realistic estimate of equilibrium power level at 2200 psi. Case C assumes the same power level but includes a longer time for corrective actions.

The results of energy balance calculations to assess the thermal response of the system are given in Table II. The conclusions drawn from these results are:

(i) - Energy generated in the transient criticality and in sustained criticalities where corrective action is effective within an hour is consumed in raising the fuel temperature and heating the RCS water. There is insufficient energy left to uncover the core.

(ii) - Most of the energy generated by the sustained criticality of longer duration must be removed from the RCS in order to avoid loss of water inventory and subsequent fuel melt (i.e., the energy cannot be absorbed within the RCS itself).

(iii) - The energy to be dissipated in order to prevent core uncovery and fuel melt is within the range of heat removal capability for natural circulation through the steam generators.

(iv) - Substantial periods of time exist prior to the calculated initiation of fuel melting should core cooling be lost.

The fission product inventories generated in the criticality events analyzed are given in Table III. The current inventory at TMI-2 is shown for comparison. The conclusions drawn from this table are:

(i) - The total inventory of fission products generated during transient criticality is insignificant relative to the current TMI inventory.

(ii) - The total inventory of fission products generated during sustained criticality is comparable to the current TMI inventory.

(iii) - All criticalities generate inventories of the volatile xenon and iodine isotopes many orders of magnitude greater than those in the current TMI inventory.

TABLE III. FISSION PRODUCT INVENTORIES GENERATED DURING CRITICALITY EVENTS

	TRANSIENT (PULSE) CRITICALITY	SUSTAINED CRITICALITY			CURRENT TMI INVENTORY
		CASE A	CASE B	CASE C	
Energy Generation Rate (MW)	2772	27.7	27.7	277	-
Fraction of Core Which is Critical	0.1	1.0	1.0	1.0	-
Time at Power (MIN)	1	60	60	600	-
Total Energy Generated (MJ)	1.7×10^4	1×10^5	1×10^6	1×10^7	-
Fission Products Generated (Ci)					
Krypton	5.1×10^3	3.2×10^4	3.2×10^5	3.2×10^6	1.0×10^5
Xenon	7.1×10^3	4.4×10^4	4.4×10^5	4.4×10^6	2.3×10^{-3}
Iodine	8.4×10^3	5.1×10^4	5.1×10^5	5.1×10^6	2.2×10^{-1}
Cesium	6.4×10^3	3.9×10^4	3.9×10^5	3.9×10^6	1.2×10^6
Others	9.0×10^4	5.3×10^5	5.3×10^6	5.3×10^7	4.0×10^7
TOTAL	1.2×10^5	7.0×10^5	7.0×10^6	7.0×10^7	4.1×10^7*

*Estimated disposition of current fission product inventory at TMI-2 is as follows:

4.4×10^4 Ci of Kr in containment
4.0×10^4 Ci in RCS water
5.0×10^5 Ci in containment sump
4.4×10^4 Ci in auxiliary building storage tanks
4.0×10^7 Ci retained primarily in fuel

To assess radiological consequences of these events, it is necessary to consider the mechanisms and the driving forces by which fission products can be transported across physical barriers on the pathway to the environment. The normal physical barriers are the fuel matrix, the fuel rod clad, the reactor coolant system boundary and the containment building. In this analysis, no credit is given for fuel rod clad as a physical barrier since most of the rods were presumed to have failed in the original accident. The principal driving forces for transport are the energy generated during and following the criticality and the fluid flows across these boundaries.

Table IV describes the applicable fission product transport mechanisms and some characteristics which help relate them to the estimated consequences of criticality. When combined with an understanding of the possible physical states of the plant, the conclusions drawn from this table are:

(i) - Once criticality has occurred, there is nothing that can be done to prevent significant additional amounts of radioactivity from entering the RCS water.

(ii) - Minimizing the fuel temperature during and following criticality will be most effective in preventing still much larger amounts of radioactivity from being available for transport.

(iii) - Maintaining isolation of the RCS while assuring core cooling is the earliest opportunity to limit the spread of radioactivity to the environment.

(iv) - Assuring the operability of the containment engineered safety features, e.g., the sprays, is an effective way to retain radioactivity inside the containment if core cooling is lost.

(v) - Maintaining isolation of the containment is the last opportunity to limit the spread of radioactivity to the environment.

TABLE IV. CONSIDERATION OF FISSION PRODUCT TRANSPORT MECHANISMS

MECHANISM	WHERE APPLICABLE	CHARACTERISTICS OF MECHANISM
SOLID STATE TRANSPORT		
KNOCKOUT	RELEASE FROM FUEL MATRIX TO RCS WATER	TYPICALLY 1-10% RELEASE INCREASES WITH FUEL FREE SURFACE AREA INDEPENDENT OF FUEL TEMPERATURE RELEASE-TO-BIRTH RATIO IS LOW AND INDEPENDENT OF FP VOLATILITY
DIFFUSION	RELEASE FROM FUEL MATRIX TO RCS WATER OR STEAM	CONTROLLED BY PRODUCT OF TIME AND FUEL TEMPERATURE RELEASE-TO-BIRTH RATIO IS HIGH AND INCREASES DIRECTLY WITH FP VOLATILITY
AEROSOL	WITHIN AND FROM RCS	VAPORIZATION AND CONDENSATION OF LOW VOLATILITY FP REQUIRES FUEL TEMPERATURES >1800C
	WITHIN AND FROM CONTAINMENT	CONTAINMENT SPRAYS REMOVE THEM EFFECTIVELY AGGLOMERATION AND SETTLING INCREASE WITH TIME REGARDLESS OF SPRAYS
LIQUID TRANSPORT	WITHIN AND FROM RCS	INCREASES WITH LEAKAGE FROM RCS
	WITHIN AND FROM CONTAINMENT	INCREASES WITH LEAKAGE FROM CONTAINMENT
VAPOR TRANSPORT	WITHIN AND FROM RCS	INCREASES WITH STEAM FLOW IN RCS INCREASES WITH HIGH WALL TEMPERATURES INCREASES WITH LEAKAGE FROM RCS
	WITHIN AND FROM CONTAINMENT	INCREASES WITH ΔP ACROSS CONTAINMENT INCREASES WITH HIGH WALL TEMPERATURES

Based on the material presented to this point, we have attempted to estimate the effects of criticality on dose rates inside containment and on release of radioactivity to the environment. These depend strongly on the efficacy of core cooling and on the pathways available for fission product transport.

The most probable sequence of events given the occurrence of criticality is that core cooling is achieved via conduction to surroundings and natural circulation through the steam generators. If cooling is efficient enough, it is likely that the system pressure can be maintained below the relief and safety valve set points, thereby maximizing RCS integrity. Given this sequence, the principal release of fission products will be from the fuel to the RCS water during fission. Assuming a 10% release fraction characteristic of knockout, sustained criticality Case B would produce a twenty fold increase in the gross radioactivity level of the RCS water (currently 4×10^4 Ci in 90,000 gal water). Of course, much larger increases in the concentrations of short-lived isotopes such as I-131 would be observed. With successful isolation of containment, releases to the environment would be controlled and possibly too low to measure.

If core cooling were deficient enough to allow relief valves to open or if a lower pressure path from the RCS to containment were available, RCS fluid would leak out taking with it noble gases and some dissolved and particulate radioactive material. Leaked water would enter the containment sump, and the noble gases would increase the radioactivity levels in containment. Emptying the entire inventory of the RCS (7.4×10^5 Ci in 90,000 gal) into the containment sump (5×10^5 Ci in 600,000 gal) would more than double the radioactivity contained there. This source would increase dose rates in the sump region. It would, however, little affect dose rates in the upper regions of the containment unless the containment spray recirculation system were activated.

Dose rates in the containment would increase as a result of a factor of 2.5 increase in the noble gas inventory. Naturally, these dose rates would drop as the short-lived isotopes decayed. However, the net long term effect would be a substantial increment above the current dose rates. Subatmospheric pressure in the currently isolated containment keeps noble gases from leaking out. Pressure increases could negate this effect, but the driving forces associated with this event do not appear great enough to produce significant out-leakage.

In the less probable event that core cooling were deficient enough to allow uncovery of the core, quantum increases in the amounts of radioactivity released from the fuel would be observed. This could be accompanied by a breach of the RCS boundary and dose rates in containment would certainly increase by an order of magnitude or more. Reliance for consequence mitigation would be placed on the containment and its engineered safety features.

At this point comparison with the results in Figure 1 is appropriate. The major difference between this analysis of consequences and that of Reference 1 is the assumed fission product inventory. Reference 1 assumed the current TMI-2 inventory, i.e., no increased inventory as would be produced by any of the criticality events described in Table III. Here we assume the more conservative energy release and inventory of sustained criticality Case C. Comparisons are made for the following circumstances: (i) meltdown inside an essentially intact containment and (ii) meltdown inside an unisolated containment without containment heat removal or sprays.

(i) - For meltdown within an intact containment, the conditional probability of latent cancer fatality would increase by no more than a factor of eighty; from 10^{-9} to 8×10^{-8} at five miles. At such low probability values, such a difference is insignificant, since it is within the uncertainties of the baseline value (i.e., factor of 100). The increase is attributable to the eighty-fold increase in the noble gas inventory, which is assumed to leak at one volume percent per day. The less volatile fission products, including most of the iodine, would be retained effectively by containment sprays and natural agglomeration and settling. Melt-through of the containment base mat would not occur.

(ii) - For meltdown inside an unisolated containment without containment heat removal or sprays, the conditional probability of latent cancer fatality would increase by about a factor of three; from 10^{-6} to 3×10^{-6} at five miles. At such low probability values, such a difference is difficult to distinguish. The increase is attributable to the gross inventory increase generated by the criticality (i.e., 7.0×10^{7} Ci added to the 4.1×10^{7} Ci already there) and to the presence of the volatile short-lived isotopes, all of which exit containment. The potential for thyroid nodules resulting from the release of I-131 would be roughly ten times that for latent cancer.

Therefore, even for the conservative cases assumed here, the off-site consequences as expressed in terms of probability of latent cancer fatality are negligible compared to the normal incidence of that health effect. No estimates were made here of the potential for land contamination or psychological effects. Only meltdown inside an unisolated containment without containment engineered safety features would likely result in significant land contamination.

It is important to keep in mind when considering these results that the probability of criticality is <u>not</u> unity as has been assumed here. Nor are the probabilities of failure of engineered safety features unity. Precautions are taken to ensure that such probabilities are as low as practical for TMI. The point is that consequence analyses such as these must be taken in context with their associated probabilities.

5. CONCLUSIONS AND RECOMMENDATIONS

The conclusions of this study are as follows:

(i) - Previous studies performed independently are in substantial agreement regarding the necessary and sufficient conditions which must be met in order to achieve criticality in the TMI-2 core. This study has uncovered no evidence to the contrary. Furthermore, there has been no indication of gross inaccuracies in the findings of previous studies which would tend to underestimate the likelihood of criticality.

(ii) - Previous studies have assumed that the most probable cause of recriticality is boron dilution. This study has examined other mechanisms by which boron might be lost from the core and has reached the same conclusion.

(iii) - Previous studies have made no attempt to quantify the absolute probability that the necessary and sufficient conditions for criticality will be satisfied at TMI-2. Rather, it is believed that the most probable mechanism for recriticality, i.e., boron dilution, is a slow enough process that the approach to criticality will be detected and corrective actions taken, provided adequate instrumentation, procedures and equipment are available. This study agrees with that approach. It concludes that to the extent boron concentration in excess of 3500 ppm can be ensured, the probability of criticality is minimized.

(iv) - Given the emphasis on preventing criticality in previous studies, little attention has been paid to potential radiological consequences. Only the most recent task force report (1) considers consequences. It indicates that latent cancer risk to off-site individuals from criticality is many orders of magnitude lower than the probability of fatality from common accidents and from all causes of cancer. This study indicates that Reference 1 may have underestimated the potential consequences of criticality but not by enough to affect the basic conclusion.

(v) - The most probable direct radiological consequence of criticality is the increase in dose rates inside containment. The magnitude of this increase depends primarily on the efficacy of core cooling and the ability to maintain RCS integrity. Depending on that magnitude, the duration of the cleanup effort could be extended significantly. Increased indirect consequences such as higher occupational exposures and greater likelihood of key equipment failure might be anticipated.

(vi) - Most probably, criticality will not result in significant off-site radiological consequences. For less probable events there are sizable variations, i.e., one or more orders of magnitude within the spectrum of off-site consequences that can be calculated. The more severe consequences are less probable since they involve multiple failures of independent systems. To the extent that core cooling and containment integrity can be maintained, the off-site consequences are minimized.

Previous studies have presented many recommendations designed to reduce the risk of criticality. We have made no effort to review all of these recommendations or to inquire as to their implementation. Rather, we present here some recommendations which occurred to us during the course of this study and propose that those responsible for the operations at TMI take them under advisement.

(i) - To minimize the potential for criticality when the mini decay heat removal system is put into service, the following recommendations are made:

- All MDHRS water should be borated to 3850 ppm.
- The system should be started with low flow to facilitate mixing of the MDHRS water, the water in the lead-in pipe run, and the water in the pressure vessel.
- Boron concentration should be monitored more frequently; as frequently as practical during startup and no less than once per shift afterward. The sampling port should be within the flow and as close to the core as is practical.
- The system should be instrumented with radioactivity monitoring equipment, either gamma or delayed neutron detectors, so as to provide a diverse measure of approach to criticality.

(ii) - Review the potential for introducing unborated water into the RCS and generate administrative preventive measures where appropriate.

(iii) - Prepare procedures to guide the operators regarding corrective action should a decrease in boron concentration be detected for whatever reason.

(iv) - Prepare procedures to guide the operator in the event that instrumentation to monitor the approach to criticality is lost.

(v) - Investigate a standby neutron poison injection system to supply back-up in the unlikely event that a pocket of low boron concentration should be swept into the core. Chemical compatibility of boric acid with cadium nitrate or sulfate or gadalinium nitrate should be investigated as an alternate to a concentrated boric acid injection.

(vi) - Review the instrumentation available to provide direct and indirect measures of criticality and the readings likely to be observed.

(vii) - Prepare procedures to guide the operators regarding corrective action should criticality be detected for whatever reason.

(viii) - Have procedures and equipment available for ensuring and confirming heat removal through the steam generators.

(ix) - Review the capabilities and procedures for isolating the RCS in its current configuration.

(x) - Review the capabilities and procedures for operating containment engineered safety features and for isolating containment.

(xi) - Place high priority on augmenting the excore neutron monitoring capability once containment entry has been gained.

(xii) - Repeat the review of recriticality prior to removal of the reactor vessel head to take into account new information.

REFERENCES

1. Haller, N.M., et al, "Evaluation of the Cleanup Activities at Three Mile Island, " USNRC, February 28, 1980.

2. Memorandum, R. Bernero (NRC/RES) to R. DiSalvo (NRC/RES), "Special Assignment - TMI-2 Hazard Evaluation," March 10, 1980.

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

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

5. Memorandum, H. J. Richings (NRC/NRR) to R. Mattson (NRC/NRR), "TMI-2 Event Excore Neutron Detector Readings: Cause and Significance," August 24, 1979.

6. Letter, G. Kulynych (B&W) to R. Harding (Metropolitan Edison Co.), "Basis for Tech Spec Boron Limits," May 1, 1979.

7. Memorandum, D. Cokinos (BNL) to D. J. Diamond (BNL), "Recriticality Calculations for TMI," Brookhaven National Laboratory, May 18, 1979.

8. Memorandum, G. R. Bond (GPU Service Company) to B. D. Elam (GPU Service Company), "Recommended Boron Concentration Levels in TMI-2," August 8, 1979.

9. Barr, E. W., et al, "TMI-2 Post-Accident Criticality Analysis," TDR-049, General Public Utilities Service Company, August 31, 1979.

10. Westfall, R. M., et al, "Criticality Analyses of Disrupted Core Models of Three Mile Island Unit 2," ORNL/CSD/TM-106, Oak Ridge National Laboratory, December 1979. (Prepared for the President's Commission on the Accident at Three Mile Island).

11. Memorandum, C. R. Marotta (NRC/NMSS) to N. M. Haller (NRC/MPA), "Nonconservative Reference Keff of TMI-2 Core as Given in Special Task Force Report 'Evaluation of the Cleanup Activities at Three Mile Island,' Dated February 28, 1980," April 1, 1980.

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APPENDIX A

SUMMARY OF CRITICALITY ANALYSES OF TMI-2

APPENDIX A

SUMMARY OF CRITICALITY ANALYSES OF TMI-2

NRC Staff Report - "Evaluation of Long-Term Post-Accident Core Cooling of Three Mile Island Unit-2," NUREG-0557, April 1979.

This analysis is based primarily on calculations by the B&W naval criticality group using the KENO-IV Monte Carlo code. Calculations and cross sections have been tested against many experiments. The calculations on slumped cores assume no control rod or burnable poison material in the core.

The major conclusions of this analysis are:

1. For no collapsing of "layers" the system is will subcritical at a boron concentration of 1500 ppm.
2. For a collapse of 3 "layers", giving a combination of about 42% of the reactor fuel, criticality would be approached at 1500 ppm but it would be about 4% subcritical at 2200 ppm.
3. For a collapse of 5 "layers", giving a combination of about 71% of the reactor fuel, the system would be several percent supercritical at 2200 ppm but several percent subcritical at 3000 ppm.
4. For a complete combination of all fuel, either in a cylinder or sphere the system would be slightly subcritical at about 3000 ppm.

This last result is the basis for B&W advocating a boron level of 3000 ppm to cover the most extreme configuration.

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

Caluclations were KENO-IV Monte Carlo code.

Assumptions:

a. no control rods
b.. no burnable poison
c. two zone core, 2.96% outer zone and (1.98 + 2.64)/2% = 2.31% inner zone
d. no core barrel, 2 - foot unborated water reflector. This is 0.5% to 1% ΔK/K conservative.

Results:

Benchmark calculations on zero power, 530°F, clean, all rods out, just just critical TMI-2 core with 1500 ppm boron was within ±0.5 ΔK/K. Results on the as-built TMI-2 lattice and the lattice with fuel rearranged in the most reactive pitch are shown in Table I.

Local criticality: Keff of four assemblies, 2.96% enriched fuel in square array by pure water

(B)ppm	Keff
2500	0.839 ± 0.004
2000	0.866
1500	0.886
1000	0.924
500	0.953
0	1.000

Conclusions: 2% of core filled with pure water will result in local criticality. The technical specification boron concentration of 3500 ppm will reduce the Keff of the most reactive configuration in Table A-I to less than 0.90.

TABLE 1

K_{eff} of TMI-2 Core As Function Of PPM Boron in Water
(No Control Rods or Burnable Poisons)

(Room Temp)

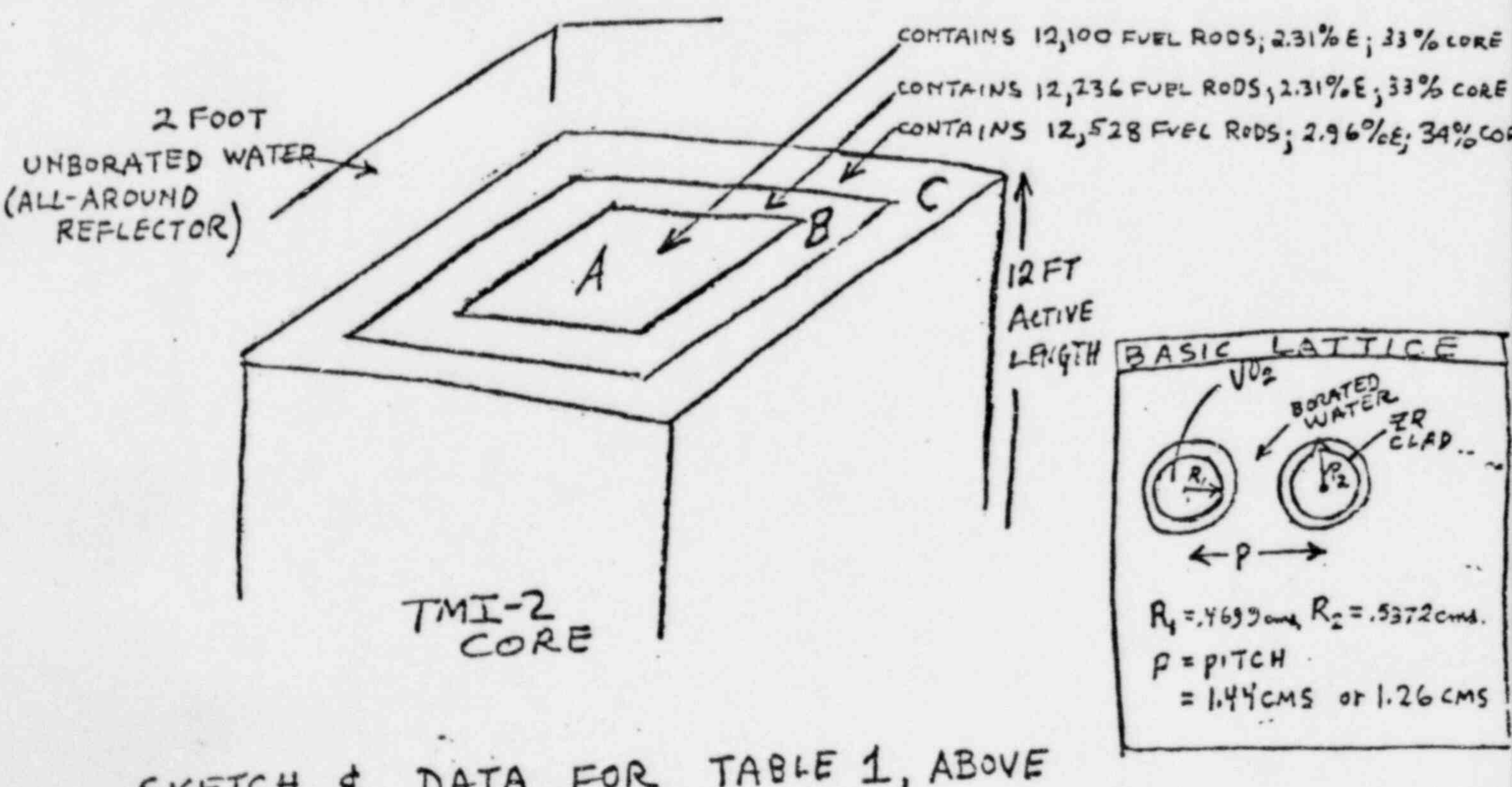
AS BUILT PITCH 1.44 cms

ZR-CLAD	A (PPM BORON)	B (PPM BORON)	C (PPM BORON)	K_{eff}*
YES	1500	1500	1500	1.040
YES	3000	3000	3000	0.883
NO	3000	3000	3000	0.857

MOST REAC PITCH 1.26 cms

ZR-CLAD	A (PPM BORON)	B (PPM BORON)	C (PPM BORON)	K_{eff}*
YES	3000	3000	3000	0.944
YES	3000	3000	2000	0.954
YES	3000	3000	1500	0.989
YES	3000	3000	1000	0.992
NO	3000	3000	3000	0.936
NO	2500	2500	2500	0.977
NO	3000	2500	2000	1.000

*All K_{eff} calc. by KENO-123 Gps, using 15,000 neutron histories and all within ± 0.004 in K_{eff} for 1 St.dev.



SKETCH & DATA FOR TABLE 1, ABOVE

Letter, G. F. Kulynych (B&W) to R. W. Harding (Metropolitan Edison Company), "Basis for Tech Spec Boron Limits," May 1, 1979.

B&W recommends a lower limit of 3000 ppm based on B&W criticality calculations:

B&W recommends an upper limit based on solubility of

Temperature ^{o}F	(B)ppm
40	4000
50	5000
60	6000
70 and higher	7000

The material attached to the letter gives the following data on the undamaged core at 88.3 EFPD at cold shutdown from PDQ-7 calculations.

T, ^{o}F	Control rods	Keff	Boron, ppm
70*	all out	.95	2155
70*	all out	.99	1795
70*	all in	.95	1705
70*	all in	.99	1385
280*	all out	.97	2100
280*	all in	.92	2100

* No credit for Lumped Burnable Poison, no Xe^{135},
No credit for Sm buildup, 1% $\Delta K/K$ conservation.

Memorandum, D. Cokinos (BNL) to D. J. Diamond (BNL), "Recriticality Calculations for TMI," Brookhaven National Laboratory, May 18, 1978.

This analysis used the HAMMER multigroup, integral transport theory code. This code, originally developed by duPont at Savannah River Laboratory and revised by EPRI-NP-565 in October 1978, has been successfully used by the nuclear industry for years. It offers an analysis of TMI-2 criticality that is completely independent from the ORNL and NRC/NMSS Monte Carlo (KENO) calculations:

The cases considered were pellet slump with no control rod or burnable poisons based on the average fuel enrichment of 2.6%. The results are:

% core slumped	critical Boron concentration
30%	2720 ppm
50%	2900 ppm
100%	3060 ppm

Memorandum, G. R. Bond (GPU Service Company) to B. D. Elam (GPU Service Company); "Recommended Boron Concentration Levels in TMI-2," August 8, 1979.

This internal memorandum recommends TMI-2 boron concentrations based on calculations later reported in Reference 9.

Recommendations:

Minimum Boron Concentration	3500 ppm
Target Boron Concentration	3900 ppm
Maximum Boron Concentration	4300 ppm

The basis of these recommendations is the analysis of the following configurations:

Configuration	1% Shutdown Boron Concentration
1. Optimum Pellet Water Density Mixture	3400
2. Total Fuel Pellet Slump (SLAB)	3470
3. Intact High Enrichment Fuel, No Discrete Poison Control	3270

The target and minimim concentrations are a direct result of the current evaluation. The maximum concentration is unchanged from the proposed TMI-2 Technical Specifications and remains substantially below the theoretical boron solubility limit. Consequently, no significant boron precipitation is expected.

Barr, E. W., et al, "TMI-2 Post-Accident Criticality Analysis" TDR-049 GPUSC Nuclear Analysis Section, August 31, 1979.

Calculations presented in the analysis were made by two different methods. Monte Carlo Calculations with the KENO-IV code, the same code used by B&W, ORNL and NRC/NMSS and the XPOSE computer code. XPOSE is an Exxon Nuclear Company version of the widely used Westinghouse LEOPARD code. Both have been widely checked against experiments and approved by NRC for licensing purposes.

The conclusions of this study were previously summarized in Reference 8. A careful review of the details of the calculations on models in this 100 page report leads to the conclusions that the recommendations of Reference 8 are valid.

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

This study was prepared for the President's Commission on the Accident at Three Mile Island. Two Monte Carlo codes were used for this analysis; KENO IV and MORSE-SGC/S. A 27 group cross section library was used which is a subset of an 218-group ENDF/B-IV library. The calculational methods were checked against 7 critical experiments and the TMI-2 startup criticality tests. In the range of water-metal ratios of interest, the calculations underpredict Keff by about 1.3% ΔK/K. An adjustment for this was made when using these results.

Three models of core disruption were studied. All cases had 3180 ppm boron in the water. These are:

a. MORSE-SGC/S Three Jump Slump Core Model as shown on Figure 1. (Note the figure and table number from the original report). Results are given in Table 13. Correcting Case B for the effect of burnable poisons (LBP) and the 1.3% ΔK/K bias:
$$0.875 + 0.006 + 0.013 = 0.894$$

b. KENO-IV Displaced Fuel Slump Model as shown in Figure 2. Results are given in Table 14. Correcting Case B for the 1.3% ΔK/K bias:
$$\text{Keff} = 0.870 + 0.013 = 0.883$$

c. KENO-IV In-Place Fuel Slump Model as shown in Figure 3. Results are given in Table 15. Correcting the 50% swelling case which has the highest Keff for the 1.3% ΔK/K bias:
$$\text{Keff} = 0.845 + 0.013 = 0.858$$

The present Technical Specifications on boron is 3500 ppm while these calculations were at 3180 ppm. The additional 320 boron would lower Keff by about 3%.

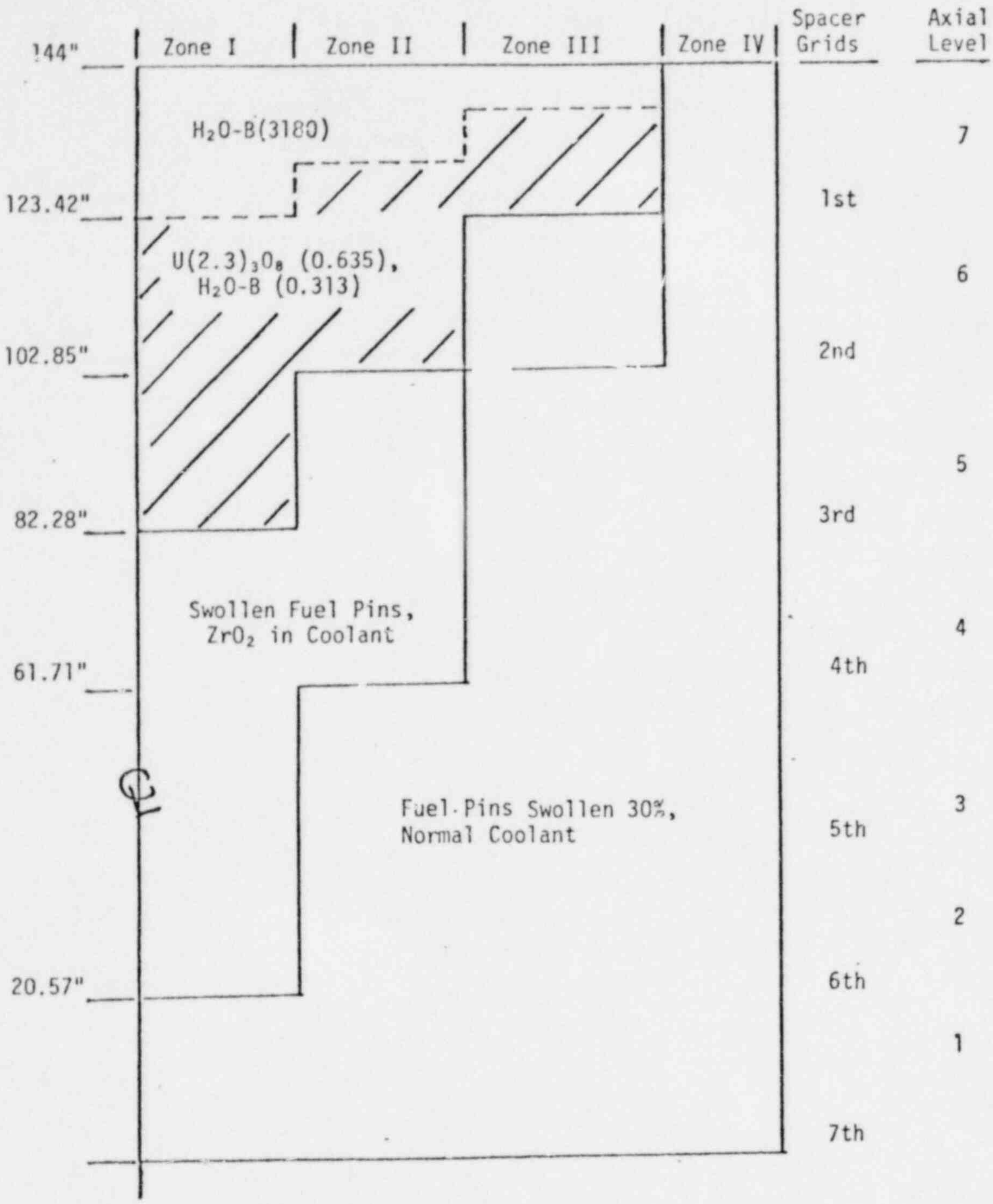


Fig. 1. MORSE-SGC/S Three Jump Slump Core Model*

*Control and Lumped Burnable Poison Rods from Disrupted Portion of Core Missing. Boron in Coolant in All Zones at 3180 wppm. Core Barrel, Radial, and Axial Reflector Regions in Model.

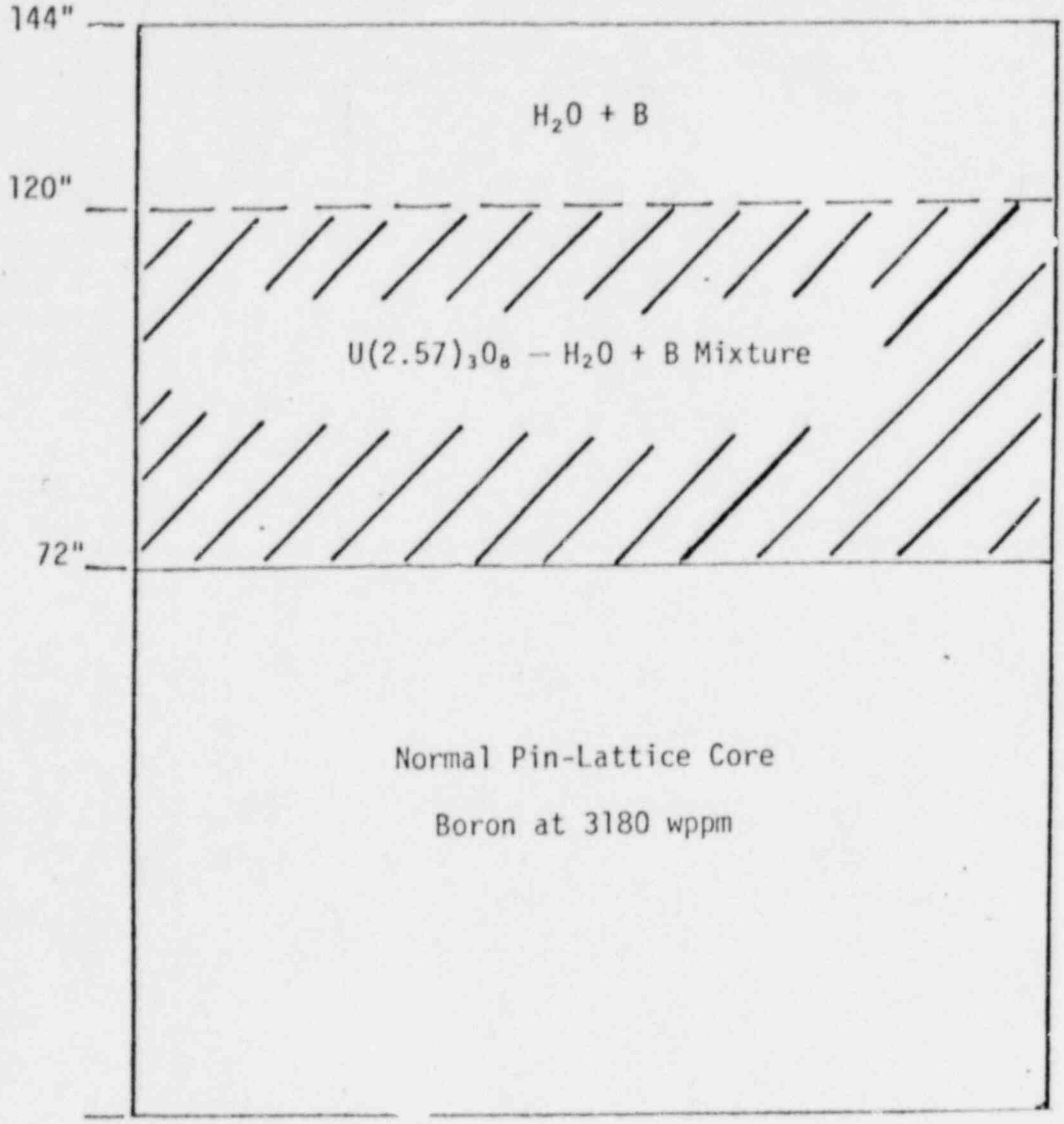


Fig. 2. KENO-IV Displaced Fuel Slump Model*

*Includes Radial and Axial Reflectors of $H_2O + B$

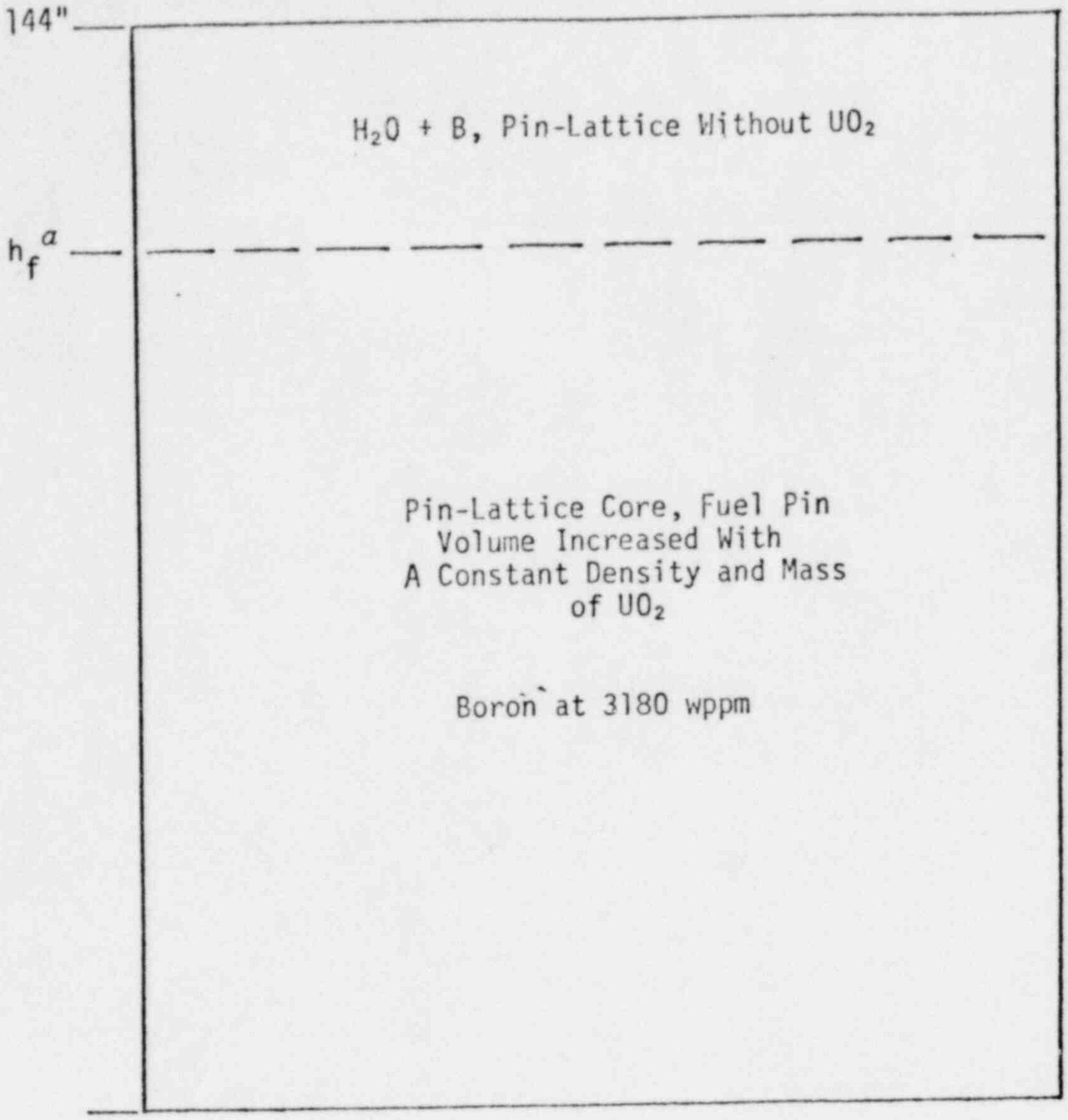


Fig. 3. KENO-IV In-Place Fuel Slump Model[b]

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[a] h_f values: 144", 114.2", 94.6", 80.8", 70.4"

[b] Includes Radial and Axial Reflectors of H_2O + B

Table 13. MORSE-SGC/S "Three Jump Slump" Disrupted Core

Case Description	Multiplication Factor
A. Base configuration[a]	0.862 ± 0.006
B. Case A with control rods out	0.875 ± 0.006
C. Case A with LBP rods removed	0.868 ± 0.006
D. Case A with controls rods and boron[b] out	1.079 ± 0.012
E. Case A with LBP rods and boron[b] out	1.043 ± 0.010
F. Case A with control rods inserted, boron out	0.988 ± 0.011

[a]13.5% of upper middle core collapsed as U_3O_8-H_2O mixture; ZrO_2 distributed in coolant channels of lower core; intact portion of fuel pin swollen by 30%; boron in coolant at 3180 wppm.

[b]Boron remaining in U_3O_8-H_2O mixture.

Table 14. KENO-IV "Displaced-Fuel Slump" Disrupted Core[a]

Case Description	Multiplication Factor
A. Base configuration	0.845 ± 0.006
B. Case A with control rods out	0.870 ± 0.006
C. Case A with boron out[b]	1.080 ± 0.006

[a]Upper 50% of core collapsed as U_3O_8-H_2O mixture; corresponding portions of control and LBP rods missing; lower half of core in normal configuration; boron in coolant at 3180 wppm.

[b]Boron remaining in U_3O_8-H_2O mixture.

Table 15. KENO-IV "In-Place Fuel Slump" Disrupted Core[a]

Assumptions: Fuel stays at constant density (0.925 of theoretical); Zr clad expands at constant volume;[b] fuel height drops to conserve volume.

Swelling (% of Max)	Height (cm)	Fuel OD (cm)	Clad OD (cm)	Min. Gap between pins (cm)	KENO-IV k-eff[c]	XSDRNPM[d] Lattice k_∞
None	365.8	0.94	1.092	0.176	0.737±0.006	0.907
25%	290.0	1.056	1.179	0.132	0.807±0.006	0.980
50%	240.2	1.160	1.273	0.085	0.845±0.005	1.014
75%	205.2	1.255	1.360	0.042	0.840±0.006	1.005
100%	178.8	1.344	1.443	0.0	0.812±0.0073	0.950

[a] Boron at 3180 wppm, constant lattice pitch = 1.443 cm.

[b] Constant clad volume, interior radius increases.

[c] Clad, control rods & LBP rods above, core as normal.

[d] 2.57 wt % enriched UO_2 (core average).

APPENDIX B

LOW BORON CONCENTRATION AT TMI-2

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

MAR 28 1980

Docket No. 50-320

MEMORANDUM FOR: R. DiSalvo, Probabilistic Analysis Staff, RES

FROM: A. J. Ignatonis, NRC/TMI Technical Support Staff

REFERENCE: B&W Letter TM/235 LCR/171 from L.C. Rogers of B&W to Messrs. J.C. Devine and G.R. Skillman of GPU, dated February 19, 1980

SUBJECT: LOW BORON CONCENTRATION AT TMI-2

Per your request I performed some work regarding the existence of possible low boron concentration water in the stagnant loops that are connected to the Decay Heat Removal System (DHRS).

According to the Burns and Roe drawings the volume of RCS water contained between valve DH-V4B and the check valve CF-V5B (discharge side of DHRS) is estimated to be approximately 950 gallons. The volume in the drop line (suction side of DHRS) located between the intersection of the hot leg and valve DH-V3 is estimated to be approximately 430 gallons. Both of these loops are stagnant and haven't been borated since the accident. Recent information from the licensee shows that the boron concentration in these loops ranged from 1,500 ppm to 2,250 ppm prior to the accident.

I have performed a rough evaluation to determine the overall result when adding low boron concentration water to the RCS. For simplicity, complete mixing and 0 ppm boron concentration in the stagnant loop was assumed. Based on the B&W figures provided in the above reference, the boron concentration of the mixture is 3,564 ppm when mixing the following volumes: reactor vessel water (30,150 gal. @ 3,800 ppm) plus the DHRS water (1,122 gal. @ 2,270 ppm) plus the MDHRS water (200 gal. @ 0 ppm) plus the two loops--suction and discharge to the DHRS (950 gal. + 430 gal. @ 0 ppm). Also, with regard to the B&W analysis on low boron concentration provided in the above reference, the addition of the two unaccounted volumes of water would not decrease the boron concentration below 3,000 ppm during injection.

Furthermore, since there may not be complete mixing in the RCS, and there may be some other uncertainties such as cold water in stagnant lines, for safety reasons plans are being made to drain and borate the DHR and the MDHR systems prior to MDHR system operation.

If you have any further questions, you can contact me or anyone on the TMI Technical Support Staff.

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