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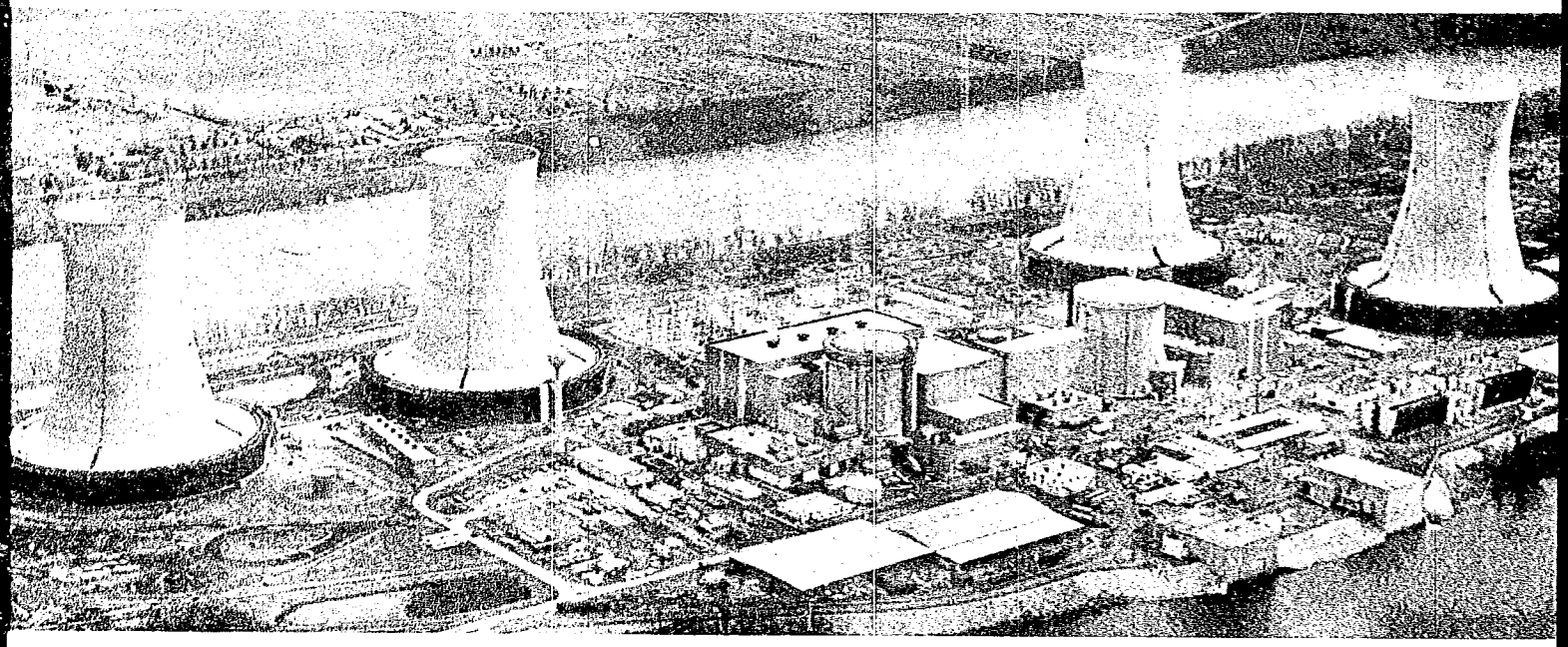

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General Public Utilities • Electric Power Research Institute • U.S. Nuclear Regulatory Commission • U.S. Department of Energy

Accountability Study for TMI-2 Fuel

Paul Goris
Dean D. Scott

May 198[illegible]

Prepared for the
U.S. Department of Energy
Three Mile Island Operations Office
Under DOE Contract No. DE-AC07-76IDO1570

Printed in the United States of America

Available from
National Technical Information Service
U.S. Department of Commerce
5285 Port Royal Road
Springfield, VA 22161
NTIS Price Codes: Printed Copy A07
Microfiche A01

GEND-016
Distribution Category: UC-78
TMI Supplement

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Published May 1981

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P.O. Box 1970
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Prepared for
EG&G Idaho, Inc..
Under Subcontract No. K-9032
And the U.S. Department of Energy
Three Mile Island Operations Office
Under DOE Contract No. DE-AC07-76IDO1570

ACKNOWLEDGEMENTS

Consulting Services

Science Applications Inc., Palo Alto, CA, particularly in the areas of nondestructive assay and weighing.

A. Sugarman
H. Bernatowicz
T. Gozani
J. Maly
R. Fullwood

Technical Conferences

Hanford Engineering Development Laboratory's Analytical Chemistry Group, Richland, WA,--Dissolution of Uranium and Plutonium in the Presence of Zirconium, Mass Spectrometry, Burnup.

M. Burt
M. Goheen
W. Matsumoto

Battelle Northwest, Richland, WA--Burnup, Isotope Correlation Studies.

C. Timmerman

Los Alamos Scientific Laboratory, Los Alamos, NM--Dissolution of Uranium and Plutonium in the Presence of Zirconium, Triple Spike Isotope Dilution Analysis.

J. Rein
R. Abernathy
J. Dahlby

Exxon Nuclear Idaho Co., Idaho Falls, ID--Burnup, Separation Chemistry for Isotope Dilution Analysis, Irradiated Uranium Material Balance Procedures.

W. Maeck
W. Emil
F. Spraktes

The authors are also grateful to Mrs. E. W. Newton for her devotion to typing and assembling the heavily formulated text; and to Mr. A. W. DeMerschman, Manager, HEDL Safeguards and Materials Management, who pursued the project as one particularly suited to HEDL experience in nuclear materials accounting.

ACCOUNTABILITY STUDY FOR TMI-2 FUEL

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SUMMARY

Material control and accountability procedures for nuclear reactors are regulated by 10CFR70.51 (Appendix 1) which provides for item accountability of the SNM entering and leaving the reactor. Because of the potential damage sustained by the TMI-2 core and the possible loss of identity of some fuel assemblies, it will not be possible to apply the requirements of 10CFR70.51 to complete recovery of the core.

A wide possible range of core damage (see Appendix 2) suggests that fuel can be grouped into a number of categories which are useful in evaluating accountability techniques. Estimated uranium fuel quantities appear in parenthesis:

- Breached Fuel rods (5.9×10^4 kg U). A broad category encompassing fuel rods with breached cladding to rubble which has deposited on the bottom of the fuel assembly.
- Unbreached Fuel Rods (1.9×10^4 kg U). Severe oxidation may have been sustained by the cladding, but no loss of fuel had occurred.
- Dispersed Fuel Material (Up to 5.0×10^3 kg U). Fuel material which has been transported by the coolant and deposited at locations in the reactor coolant system or fallen on the bottom of the pressure vessel.
- Waste (up to 500 kg U). Contamination on tools and reactor components which will be packaged and treated as waste. Will also include uranium dissolved and suspended in the coolant.
- Unrecoverable (?). Material and contamination in the reactor coolant system that has been detected, but cannot be immediately removed. Severe corrosion or diffusion reactions between the fuel material and steel in the coolant system components may be responsible for this. At certain locations in the coolant system the small quantity of fuel material may not warrant the extraordinary effort required to remove it.

The actual core damage will not be known until recovery operations begin. Therefore, accountability planning must be quite broad to cover a multitude of possible damage conditions.

Basic relationships in TMI-2 accountability are: core recovery total fuel weight inventory difference (ID), total uranium ID, U-235 ID, plutonium balance, and corresponding limits of error. The ID expresses the difference between the initial core loading and the post accident measured values. The limit of error (LEID) estimates the uncertainty range of the ID. Since plutonium was not present in the initial core loading, a plutonium balance and an associated limit of error are identified.

Weighing is recommended as a bulk measurement technique since a comparison of initial vs post accident core weight including structural components will be the first indication of recovered fuel balance. Weighing can also serve as a primary measurement method in conjunction with nondestructive assay (NDA) and chemical analysis to indicate specific SNM balances.

Active and passive NDA may also serve as bulk measurement techniques. The principal application will be measurement of SNM content directly in irregular or breached fuel assemblies.

LEID evaluations based on estimated uncertainties were determined as follows: (Listed techniques are the basic methods assumed to be applied to breached fuel):

Measurement Technique	LEID	
	Kg U	% Initial Inventory
Weighing	632	0.8
Passive NDA	1119	1.3
Active NDA	1119	1.3

Item accountability, as provided under 10CFR70.51, for intact fuel assemblies removed from a reactor, will greatly simplify accountability and reduce the range of uncertainty to the extent that it can be applied to undamaged fuel assemblies.

Chemical verification of PWR reactor code predictions based on a reasonable number of selected samples is recommended to support predicted SNM values for accountability purposes. Statistical sampling and analysis to establish core average SNM values may not be feasible due to the range of burnup levels and variations in initial U-235 abundance. Other disadvantages are core recovery impact and laboratory costs.

Dispersed fuel and waste assay will be concerned with particulate collected from the pressure vessel and cooling system, and solution or suspension type fuel content of the cooling water. Chemical sampling and assay along with NDA are proposed. Certain fuel forms such as particulate in difficultly accessible areas or plated fuel not recovered in a general cleanout and involving possible long-term cleanout can be estimated with the aid of process holdup type NDA instrumentation.

SNM tracking will be an essential component in TMI-2 accountability. Anticipated irregular fuel assembly sections, possible eutectic bonded debris, collected fuel rubble, and the large number of fuel samples expected to be shipped to various laboratories throughout the country require SNM certification which must accompany all fuel components in their destination following removal from the immediate reactor area.

General methods for inventory of TMI-2 fuel are indicated by Figure D1 on the following page.

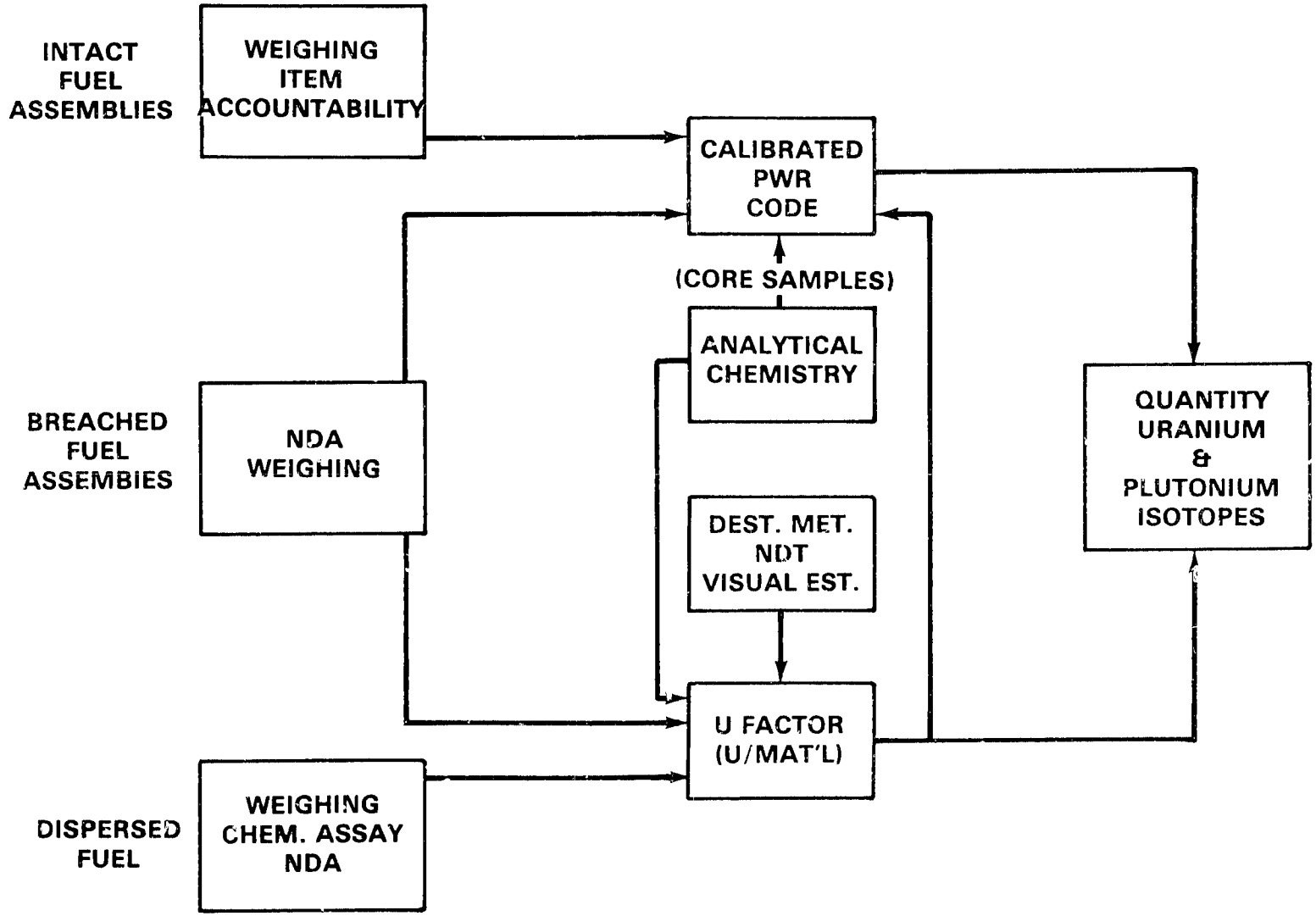


Figure DI. Methods for Inventory of TMI-2 Fuel

RECOMMENDATIONS

- Assign an accountability engineer to the core recovery planning group as soon as possible to establish mutual equipment design and procedural requirements.

- In consideration of the number of techniques which might be utilized in TMI-2 accountability, and also the problems which might be encountered; this study recommends the following general approach to TMI-2 accountability:

  1. Bulk Measurement

     - Weigh each damaged fuel assembly, and to the extent possible, the associated loose pellets, fragments, and rubble. Store all components associated with a damaged fuel assembly in an individual storage container.

     - Rely on item accountability for any undamaged fuel assemblies (or minimal damaged).

     - Weigh all pellets, fragments, or rubble not identifiable with specific fuel assemblies as removed from the pressure vessel or cooling system. Store in a minimum number of containers.

  2. Specific Measurement

     - Obtain an optimum number of fuel pellet samples from selected fuel assemblies covering the burnup range. Analyze these samples chemically for burnup, total U and Pu, and isotopic U and Pu. Compare results with PWR code predicted values for verification or calibration if necessary. Apply predicted values to item accountability, and to loose SNM associated with specific fuel assemblies. Utilize Triple-Spike Isotope Dilution Analysis.

  3. Solution, Plated, Long-Term Recoverable SNM Throughout Cooling System

     - SNM in these categories may represent a very small fraction of the total SNM and may not justify extensive recovery and accountability efforts. Reasonable estimates can be made by means of process holdup NDA instrumentation.

(See Section D, page D-1 for discussion on Bulk and Specific Measurements).

SECTION A

INTRODUCTION

The TMI-2 accountability study considers problems of identifying, measuring, and accounting for TMI-2 fuel in the resident condition, as it is removed from the reactor, during subsequent cleanup, and during post-removal examinations. The goal is to identify methods and procedures which will provide a verifiable material balance equating to the pre-accident balance.

TMI-2 operated for 94.6 effective full power days (EFPD) at the time of the accident. The average burnup has been estimated at 3176 megawatt days per metric ton heavy metal (MWD/MTHM) corresponding to approximately 0.321% of the total uranium or approximately 13.9% of the U-235 in the initial core loading. Burnup also represents fission of Pu-239 and Pu-241 produced during reactor operation.

TMI-2 accountability will be concerned with the following SNM quantities:

Initial Core Inventory:	82,900 kg	Total U
	2.63 w/o	U-235, core average
	2,180 kg	U-235
Predicted Final Core Inventory: (Based on ORIGEN-II calculations, see Table C3)	82,424 kg	Total U
	2.21 w/o	U-235, core average
	1,914 kg	U-235
	159 kg	Pu

Figures A1 and A2 indicate the estimated maximum and minimum damage to the TMI-2 core. The cross sectional areas of Figures A1 and A2 represent a vertical array of 177 fuel assemblies. Horizontal lines indicate elevations in feet. (See Figure C1 for fuel assembly positioning.)

A discussion on damage levels appears in Appendix 2 in plans for core dismantling.

Figure A3 shows a TMI-2 fuel assembly. The fuel section of the assembly is 12 feet in length and consists of 208 fuel rods, 16 control rods, and an instrumentation tube as indicated. Each assembly contains approximately 468 kg uranium distributed throughout 208 fuel rods each containing approximately 2-1/4 kg uranium. Under fortuitous circumstances whereby most of the assemblies are shown to be intact, and item accountability is applicable, the overall accountability program will be simplified and the uncertainty in the post accident inventory reduced accordingly.

Under maximum damage conditions, the Zircaloy cladding of the fuel rods in high burnup regions was oxidized to brittleness and fragmented causing openings in rods which enabled fuel pellets to fall out. It is also believed that a Zr-U-O eutectic formed in some fuel assemblies and this material lies as part of the rubble on the bottom of assemblies.

Figures A4 and A5 indicate the reactor cooling system. Estimates of up to 6% of the fuel may exist as fines throughout the cooling system. Further definition of the cooling system is given by Figure D6.

TMI-2 accountability will be concerned with evaluating SNM remaining in intact fuel assemblies, damaged fuel assemblies, rubble throughout the core, fines in the cooling system, solution forms of fuel, plated fuel throughout the cooling system, and waste.

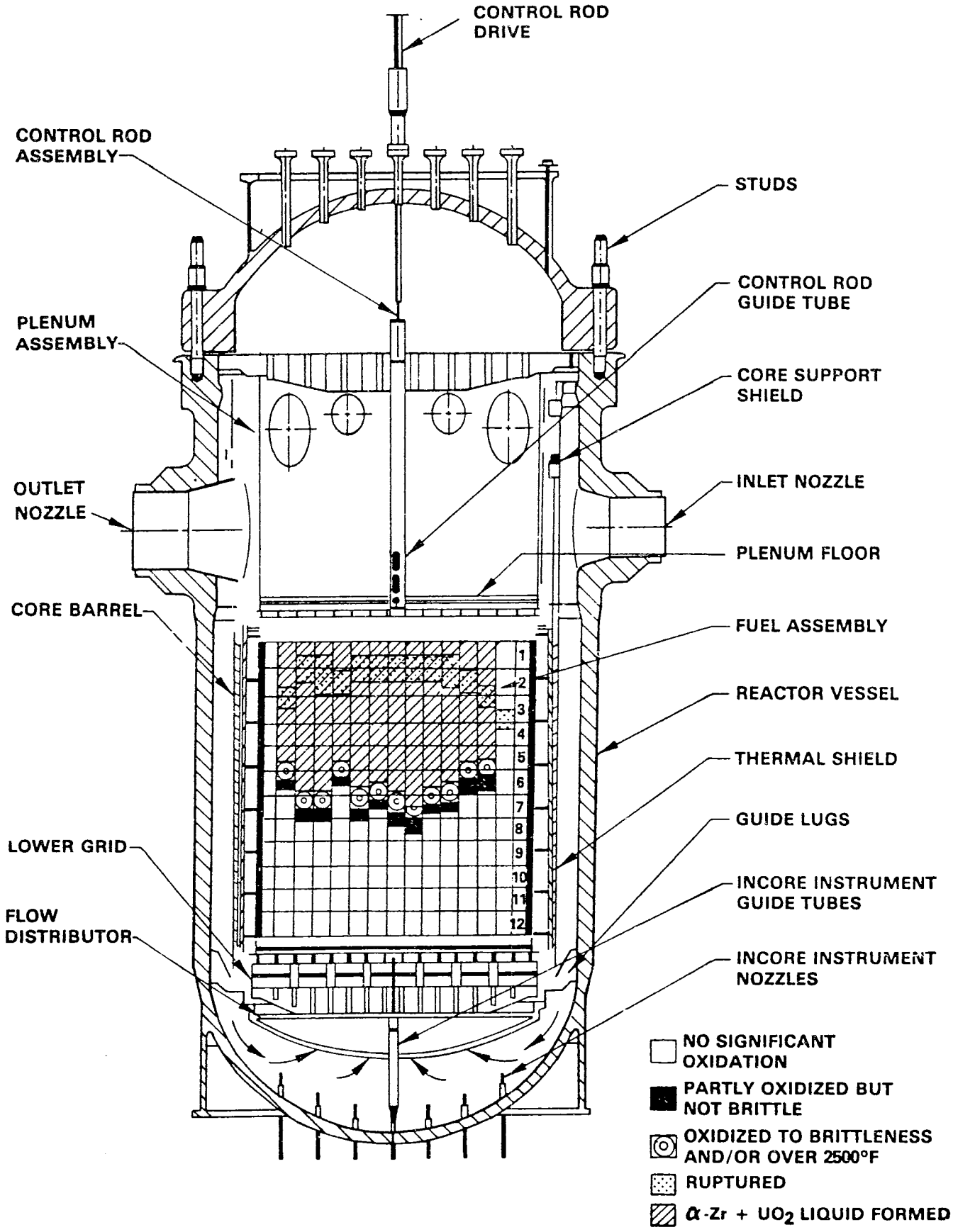


Figure A1. Estimated Maximum Damage to TMI-2 Core at Time = 3 Hours (Elevations in feet from Top of Core).

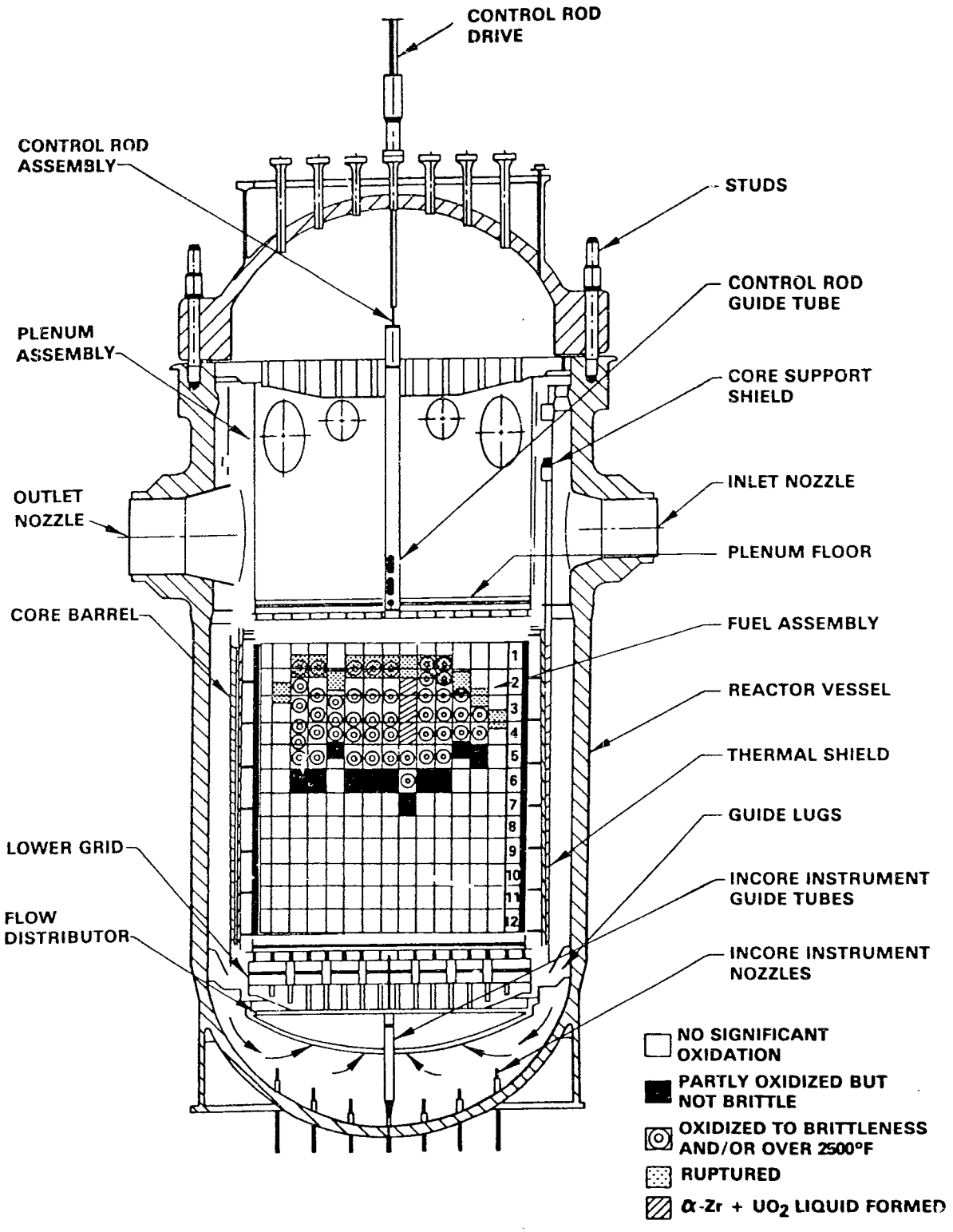


Figure A2. Estimated Minimum Damage to TMI-2 Core At Time = 3 Hours (Elevations in feet from top of core).

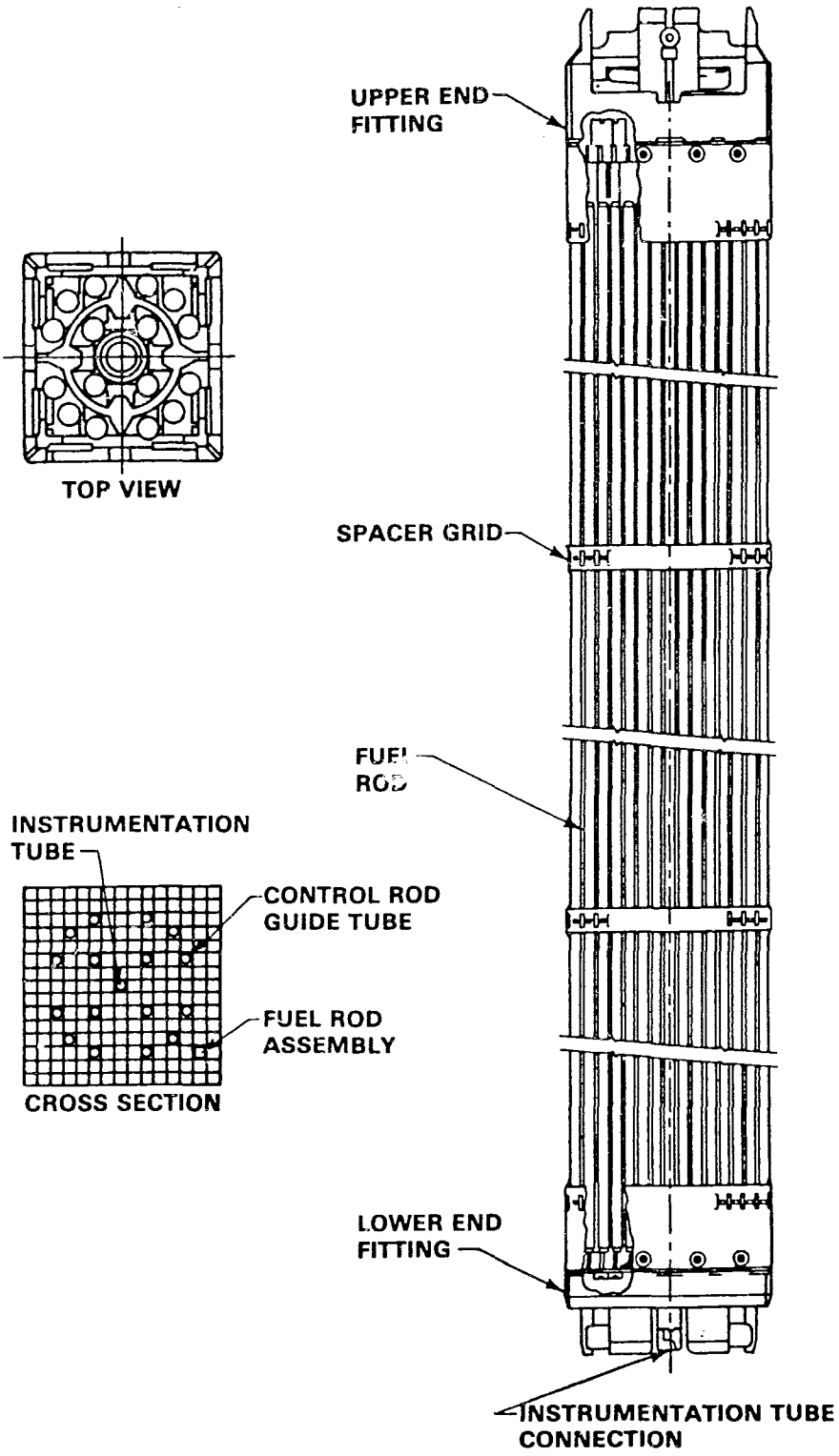


Figure A3. TMI-2 Fuel Assembly (Fuel column 144 inches).

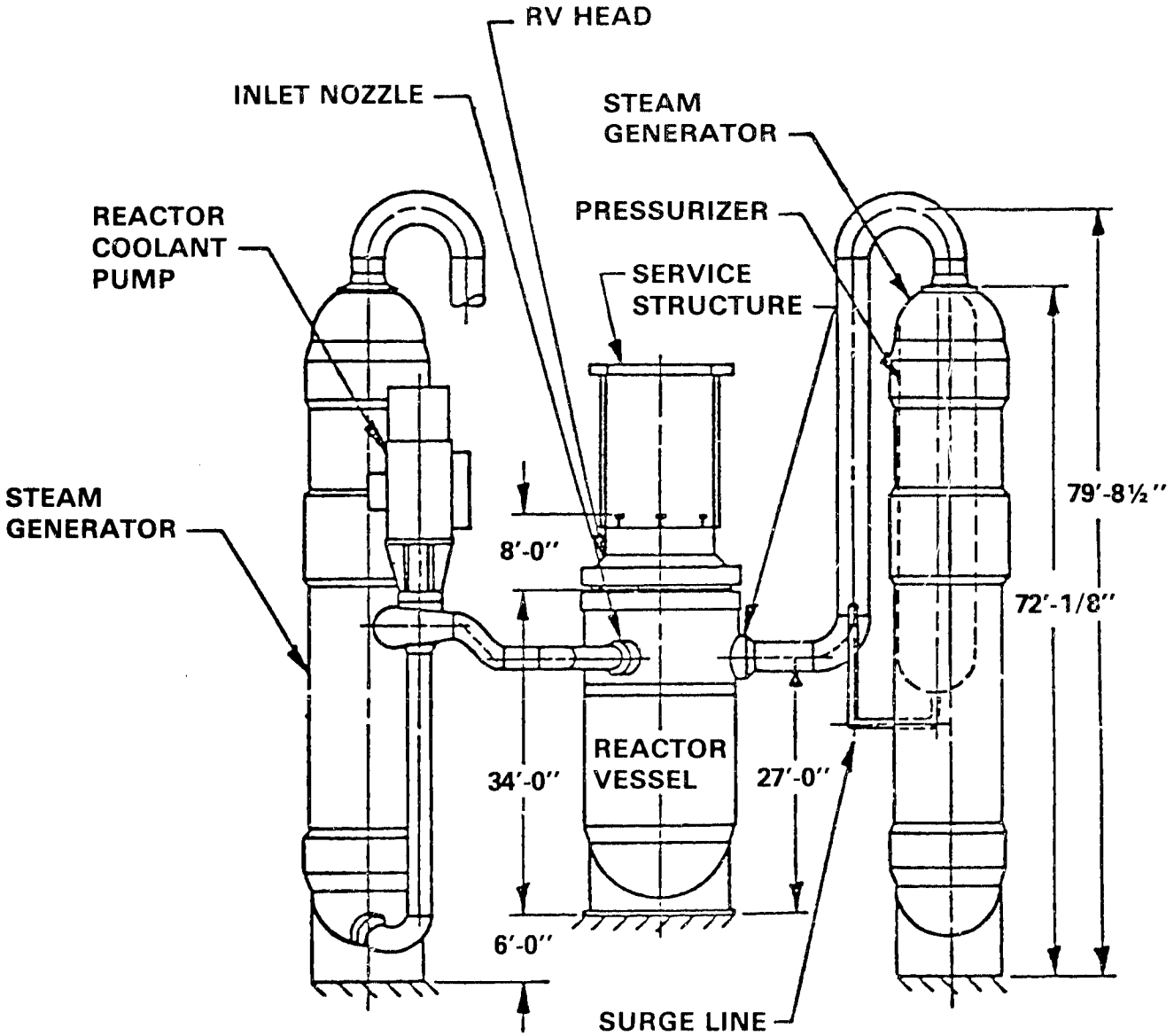


Figure A4. TMI-2 Pressurized Water System - Sectional View.

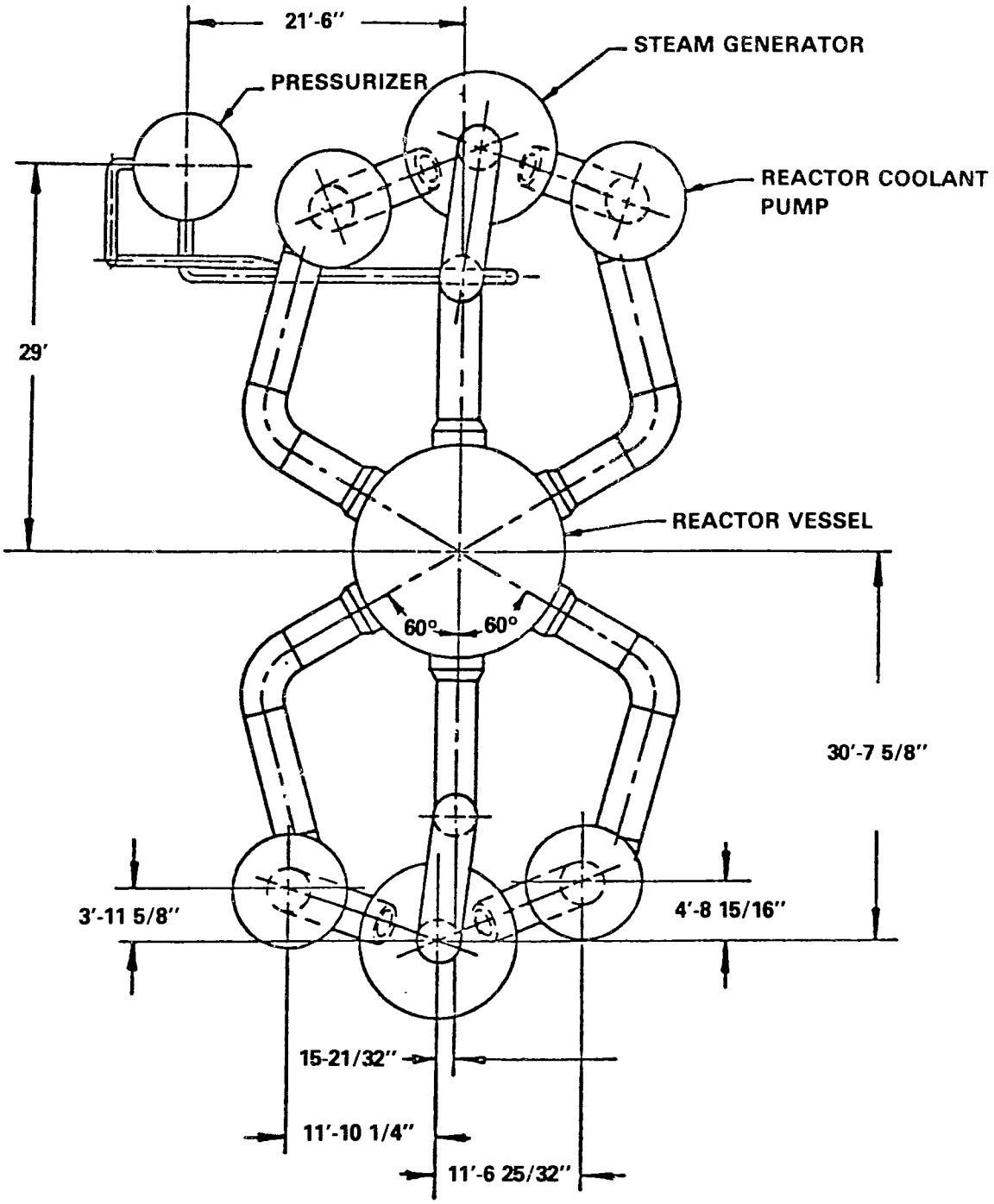


Figure A5. TMI-2 Pressurized Water System - Plan View.

SECTION B

ACCOUNTABILITY IN PREVIOUS REACTOR ACCIDENTS

A review of certain reactor accidents was carried out to assist in formulating an accountability approach in disassembly and removal of the TMI-2 core. Reactor accidents reviewed appear in Table B1. Although this review was by no means extensive, it was found that fuel accountability of the damaged core was considered only in the Stationary Low Power Plant (SL-1) accident at the National Reactor Testing Station in Idaho, January 3, 1961. A general description of the SL-1 reactor and details of the accident appear in Reference (1).

The physical operations in making an inventory of the damaged fuel from the SL-1 core consisted of weighing each fuel element, fuel plate or plate fragment remotely in the Radioactive Materials Laboratory. A Toledo scale with a 20 kg capacity was used and all weights were considered accurate to ±1%.

The uranium weights were referenced to the original loading and did not account for burnup. They were obtained by multiplying the weight of each plate by the factor .0744 to obtain the weight of U-235 per plate. For 100% recovery, the calculated recovered weight of U-235 would equal the initial loading.

Total weight of U-235 recovered in SL-1 accident:

Fuel Material Category	Grams U-235
Fuel material identifiable as to fuel plate number	11,534.9
Miscellaneous fuel material not identifiable as to fuel plate number	369.0
Reactor vessel debris	1,124.6 ± 510.0
Total U-235 accounted for	13,028.5 ± 548.0
Total Initial U-235 loading	14,007.5
Percent U-235 accounted for	93.0 ± 3.9

TABLE B1

REACTOR ACCIDENTS CONSIDERED FROM AN ACCOUNTABILITY STANDPOINT

Reactor					Accident		Fuel Damage	References
Name	Location	Type	Size	Fuel	Date	Description		
SL-1 (Stationary Low Power Plant No. 1)	National Reactor Testing Station Idaho Falls, Idaho	BWR (with natural circulation)	3000 kWt	93% enriched U-235 Al-U Plates.	1/3/61	Manual withdrawal of central control rod.	Core destroyed, vessel rose 9 ft., reactor dismantled.	1, 36
SRE (Sodium Reactor Experiment)	Santa Susana, CA	Sodium-cooled graphite moderated	20 MWt	Six foot high, S.S. clad, unalloyed, 2.8% enriched U-235	7/24/59	Coolant channel blockage by impurities, overheating, perhaps fuel bowing	12 out of 43 F/A damaged. Core removed and replaced.	36, 37
NRX	Chalk River, Canada	H_2O cooled, D_2O moderated	30 MWt	Natural unranium rods	12/12/52	Fuel rod lost coolant and overheated (perhaps local increase in flux), jet of molten Pu penetrated to D_2O	Core badly damaged, removed and replaced.	36,38
NRU	Chalk River, Canada	D_2O cooled, D_2O moderated	200 MWt	Flat, natural uranium metal bars clad in 1S aluminum	5/28/58	Accidental release of a large piece of highly active irradiated uranium into the maintenance pit of the reactor	A few uranium rods melted and damaged their containing tubes.	36, 39
WTR (Westinghouse Test Reactor)	Waltz Mills, PA	Water cooled, water moderated beryllium reflected modified MTR type	60 MWt	13% fully enriched uranium-aluminum alloy cylinders with a total element length of 3 feet	4/3/60	Fuel element failure meltdown (from lack of bonding between fuel and cladding) with subsequent blockage of coolant channel	One fuel element melted.	36, 40
EBR-1 (Experimental Breeder Reactor)	National Reactor Testing Station Idaho Falls, Idaho	Liquid-metal (NaK) cooled, heterogeneous experimental reactor	1.4 MWt	169 SS tubes that hold four, 94% enriched uranium slugs sandwiched between natural uranium	11/29/55	Core melt caused by lower than expected core power results during startup and failure to achieve a scram with first attempt when fuel temperature rise was first noticed.	Core melted, little other contamination.	41
Enrico Fermi	Monroe, Michigan	Fast Breeder, cooled by upward flowing sodium from 3 loops	200 MWt	Core is 2.56% U-235 in a U-10 w/o Mo alloy clad in 0.158 in dia., 5 mil Zr. 14700 pins in 105 subassemblies. [illegible]et is [illegible]ated U.	10/5/66	Two zirconium liners to the flow distribution core in the reactor detached and blocked the inlets to two subassemblies causing these subassemblies to fail by overheating.	Two subassemblies melted and stuck together. Some melting occurred in other subassemblies.	42, 43

SECTION C

REGULATORY REQUIREMENTS, MATERIAL CATEGORIES

10CFR70.51 provides for item accountability of SNM entering or leaving a reactor under normal reactor operations (see Appendix 1). This regulation is assumed to be applicable to TMI-2 intact fuel assemblies exhibiting insignificant to no detectable fragmentation or damage otherwise. The means for verifying such conditions, of course, may be somewhat difficult due to radiation levels. However, remote CCTV, optical viewing, ultrasonic testing, etc, should greatly assist with physical inspection of each removed fuel assembly.

The only place in the fuel cycle that transmutation occurs is the power plant. Here, the elements and isotopes of one type are converted to another type, hence, an element by element accounting is not possible. In recognition of this fact, piece by piece accountability based on the fuel serial numbers is used with the fuel fabricator's assay of the contents being accepted. In the case of physical fuel damage, such as at TMI-2, such an accounting method breaks down. In other portions of the fuel cycle, no transmutation takes place, but the SNM is no longer in discrete units susceptible to item accountability; therefore, material balance accountability is used. It is clear that the situation at TMI-2 involves aspects of both processes, i.e., transmutation and material balance accountability, a situation which regulations have not specifically addressed. This section describes regulations and guides which appear to have the most relevance to material accountability at TMI-2.

MATERIAL CONTROL AND ACCOUNTABILITY AT COMMERCIAL POWER PLANTS

10CFR70.51b (which is reproduced in Appendix 1) requires that a licensee shall keep records of receipt, inventory, disposal, and transfer of all special nuclear material in his possession regardless of its origin or method of acquisition. Comment: This requirement places the accountability responsibility on the licensee not only for the original material, but also the fission products and the capture products, such as plutonium. These records shall be kept for as long as the material is in the licensee's possession, plus five years. When the retention period is unspecified, then disposal must be authorized.

10CFR70.51e says that each licensee possessing SNM in a quantity exceeding 1 kg in activities other than a nuclear reactor licensed pursuant to Part 50 or waste disposal operations as sealed sources or as reactor irradiated fuels in research shall satisfy a number of requirements. Comment: It appears that section (e) aims at the problem of material accountability in processes not susceptible to item accountability.

Should it be agreed that because of the accident, the nuclear power plant was not operated in accordance with 10CFR50 and that 10CFR70.51e should apply, then 10CFR70.51e allows a 1% error in the LEID in a chemical reprocessing plant, or 0.5% error in other types of plants. This same section says that the ID should not exceed 200 grams of plutonium or 9 kg of low enriched uranium. These differences will be very difficult to attain at TMI-2.

10CFR70.52 requires immediate reporting of criticality accidents but is not explicit on requiring the reporting of material balances before and after the accident.

10CFR70.57 provides the requirement for establishing a measurement control program for SNM control and accounting, including calibration traceability. Presumably, the measurement and statistical procedures employed in the TMI-2 cleanup must comply with these requirements where appropriate.

A review of the Nuclear Regulatory Guides provides no information of direct relevance until Division 5.

Regulatory Guide 5.1 specifies the acceptable fuel assembly number marking requirements and procedures. Regulatory Guide 5.1 specifies acceptable procedures and methods for the nondestructive assay of scrap. The passive methods are based on: alpha, gamma-ray, and spontaneous radiation.

Regulatory Guide 5.16 applies to standard methods for chemical, mass spectrometric, spectrochemical, nuclear, and radiochemical analysis of plutonium nitrate and plutonium metal. This guide could be applicable to determining the quantity of SNM in the cooling water. It would, of course, be necessary to concentrate the material (e.g., by distillation). The residual could then be analyzed directly or dissolved in nitric acid and analyzed using methods developed for fuels reprocessing and accountability.

Regulatory Guide 5.18 addresses the acceptable limits of error and generally accepts the standard ANSI N15.16. The guide mentions 95% confidence interval measurements.

Regulatory Guide 5.19 discusses methods for plutonium nitrate solution accountability. This could be relevant to assay of the plant cooling water as well as cleanup solutions.

Regulatory Guide 5.29 addresses Nuclear Material Control Systems for Nuclear Power Plants, clarifies 10CFR70.51(c), and recommends the use of ANSI N15.8 for nuclear material control at nuclear power plants.

Regulatory Guide 5.33 addresses statistical evaluation of ID. Guide 5.34 discusses nondestructive assay of plutonium in scrap material by spontaneous fission detection using: moderated thermal neutron detectors and neutron well coincidence counters. Suitable detectors are ^{3}HeC (thermal neutrons) and ^{4}He and ZnS (for fast neutrons). Regulatory Guide 5.37 discusses in situ assay of enriched uranium residual holdup by detecting the 185.7 keV gamma ray.

Regulatory Guide 5.47 is concerned with the control and accountability of plutonium in waste material. This only seems to relate to a portion of the TMI-2 cleanup, e.g., the drummed waste, but is of little direct relevance to material on the cooling system walls, in the damaged fuel elements, or on the bottom of the reactor vessel.

Regulatory Guide 5.48 addresses the mass measurements of liquids and could directly relate to assaying material contained in the cooling water and other solutions at TMI-2.

Regulatory Guide 5.51 discusses the management review of nuclear material control and accounting systems--this could be relevant to the conduct of the cleanup.

The remaining Regulatory Guides seem to have little relevance to material accounting at TMI-2.

Table C1, based on DOE 5630.2 "Control and Accountability of Nuclear Materials, Basic Principles," ranks SNM according to possible attractiveness for diversion. (See APPENDIX 1 for complete DOE 5630.2).

Individual fuel assemblies in TMI-2 should probably be classed as Category IB material since U-235 is in combination with plutonium and the total SNM is 5000 grams or more. On the average, each assembly contains approximately 9.2 kg U-235 and approximately .90 kg plutonium according to calculations made from Table C3.

TABLE C1

RANKING OF SNM ACCORDING TO POSSIBLE ATTRACTIVENESS FOR DIVERSION

Category 1A Material

1. Assembled Plutonium (Pu* or U-233) Weapons Components
2. Assembled U-235 Weapons Components
3. Pu* or U-233 Machined Weapons Parts
4. U-235 Machined Weapons Parts
5. Pu* Metal (buttons, rods, pieces)
6. U-233 Metal
7. U-235 Metal
8. Pu* Oxides
9. U-233 Oxides
10. U-235 Oxides
11. U-235 Carbides

Category 1B Material

12. Pu-238 Oxide or Metal
13. Nitrate Crystals and Nitrate Solutions of Pu, U-233, and U-235
14. Pu, U-233, and U-235 Solutions other than Nitrate
15. Compounds of Pu, U-233, and U-235 other than those listed in items 8-12 above.
16. Pu Alloys or oxide, Carbide or Nitride Mixtures
17. U-233 Alloys or Oxide, Carbide or Nitride Mixtures
18. U-235 Alloys or Oxide, Carbide or Nitride Mixtures
19. Pu Fuel Elements and Assemblies
20. U-233 Fuel Elements and Assemblies
21. U-235 Fuel Elements and Assemblies
22. Pu, U-233, and U-235 High-Grade Recoverable Scrap
23. Pu, U-233, and U-235 in Irradiated Forms
24. Pu, U-233, and U-235 Low-Grade Recoverable Scrap (Process Residues)

*Excluding Pu-238 and Pu-242

MATERIAL CATEGORIES AND ESTIMATES

For purposes of evaluation of accountability techniques for TMI-2 fuel, the fuel damage can be placed into several categories which are described below. Different weighing methods may be required for each of the categories because of the condition of the fuel and the mass of material being measured. A summary of the quantities of material in the different fuel material categories is given in Table C2.

The largest quantity of fuel is most likely "breached" which means that the cladding has ruptured, exposing the uranium dioxide fuel to the atmosphere in the reactor pressure vessel (RPV) and little to no structural integrity remains in the fuel rods and assemblies. This condition probably applies to the inner 137 fuel rods which were the hottest at the time of the accident.

The fuel assemblies nearest the center of the core may have become so hot that the melting range of Zircaloy (close to the melting point of zirconium which is 1852°C) was exceeded and the fuel is laying as rubble on stubs of fuel rods extending up from the bottom nozzle. The estimated quantity of fuel in this breached condition corresponding to the inner 137 fuel assemblies minus the dispersed fuel is 5.9×10^4 kg uranium.

Any unbreached fuel rods are probably in the 40 peripheral fuel assemblies. Severe oxidation of the cladding in the peripheral assemblies most likely occurred, but structural integrity of the cladding and fuel assemblies is probably good. Weighing these assemblies for accountability may be relatively easy. The estimated quantity of fuel in the unbreached condition corresponding to the peripheral fuel assemblies is 1.9×10^4 kg uranium.

A small quantity of the highly damaged fuel may exist in the form of fine particles. Since the reactor coolant pumps were operated for a period of time after damage to the core occurred, some of the particulate material could have been carried in suspension by the flow of coolant and deposited in the coolant piping system. It is estimated that 1% of the core inventory

TABLE C2

ESTIMATED TOTAL URANIUM QUANTITIES IN TMI-2 FUEL CATEGORIES

Breached Fuel Rods	5.9×10^4 kg U[a]
Unbreached Fuel Rods	1.9×10^4 kg U[b]
Dispersed Fuel Material	5.0×10^3 kg U[c]
Waste	4.8×10^2 kg U[d] (3 to 4 kg U)
Unrecoverable or long-term recoverable	$\cong 0$

(a) Fuel from 137 interior assemblies

(b) Fuel from 40 peripheral fuel assemblies

(c) Quantity of particulate fuel material less than .06 cm in diameter

(d) Quantity calculated for uranium in a saturated solution of borated water in suspension. Number in parentheses is an estimate based on engineering judgment. Recent communications with Sandia Laboratories[2] where coolant sampling and analysis has been performed indicate that even the lower number may be too high.

(Appendix II.7 of Reference 3) has an average diameter of 0.24 cm. However, Powers[3] has calculated that fuel particles smaller than 0.6 cm diameter could have been carried by the flowing coolant during the period that the reactor coolant pump was operating.

A worst case estimate of the volumetric fraction of fine fragments has been made by D. A. Powers (Appendix II.7 of Reference 3) in which it was found that 6% of the total core inventory may exist as fines less than .06 cm in diameter. Therefore, the quantity of dispersed material could be .06 x 82.9 x 10^3 kg (initial core inventory of uranium) $\simeq$ 5000 kg uranium dispersed in the coolant system.

The unrecoverable fuel is material which may reside in the reactor at locations which are inaccessible and are not removed even after decontamination. Sources of this material could be fuel material which has reacted with the surfaces it has deposited on and was removed by decontamination, and fuel material which has lodged in coolant system components and cannot be removed. It is difficult to estimate the quantity of this category of fuel material, but it is expected to be very small and to contribute little if anything to the materials balance.

The inventory of uranium isotopes, plutonium isotopes, and fission products has been calculated for the TMI-2 fuel on an assembly by assembly basis (assuming 1/4 core symmetry)[4] using PDQ diffusion code and on a core average basis using ORIGIN-II code.[5] The PDQ computer code output for the isotopic and fission product concentrations on a fuel assembly basis is given in Appendix 3.

A summary of the ORIGIN-II inputs and outputs for the isotopic concentrations on a core average basis is given in Table C3. The isotopic concentrations (especially on a fuel assembly basis) will be important in calibrating NDA techniques for analyzing the spent fuel, and in confirming the results obtained from bulk measurements and analytical chemistry techniques.

TABLE C3

REACTOR DEPENDENT INPUT AND OUTPUT FOR TMI-2 ORIGEN CALCULATIONS[5]

ORIGEN Inputs

Thermal Power (Gross)	2772 MWt
Average Fuel Burnup at Time of Accident	3176 MWD/MTHM
Initial Core Loading	82.9 MTU
Average Fuel Enrichment	2.63 w/o U-235[a]

Isotope	Initial Core Inventory, grams	Final Core Inventory, grams[b]
U-238	8.070×10^7	8.0866×10^7[c]
U-235	2.180×10^6 (2.63 w/o)[d]	1.8252×10^6 (2.21 w/o)
U-234	2.073×10^4	1.974×10^4
Pu-239	0.0	1.4039×10^5
Pu-240	0.0	1.096×10^4
Pu-241	0.0	2.323×10^3
Pu-238	0.0	---
Pu-242	0.0	8.4×10^1
Total	8.290×10^7	8.26×10^7

(a) The distribution of enrichments in the original core is 56 fuel assemblies with 1.98 w/o U-235, 61 fuel assemblies with 2.64 w/o U-235, and 60 fuel assemblies with 2.96 w/o U-235.

(b) Inventory at shutdown. Data obtained from Reference 34.

(c) Difference between initial core inventory of U-238 and sum of final core inventory of actinides.

(d) 2.56 w/o U-235 is the initial average enrichment identified in, "Three Mile Island - Unit 2, Final Safety Analysis Report."

Figure C1 shows distribution of fuel assemblies at the U-235 enrichment levels and the location of B_4C lumped burnable poison at three enrichment levels.

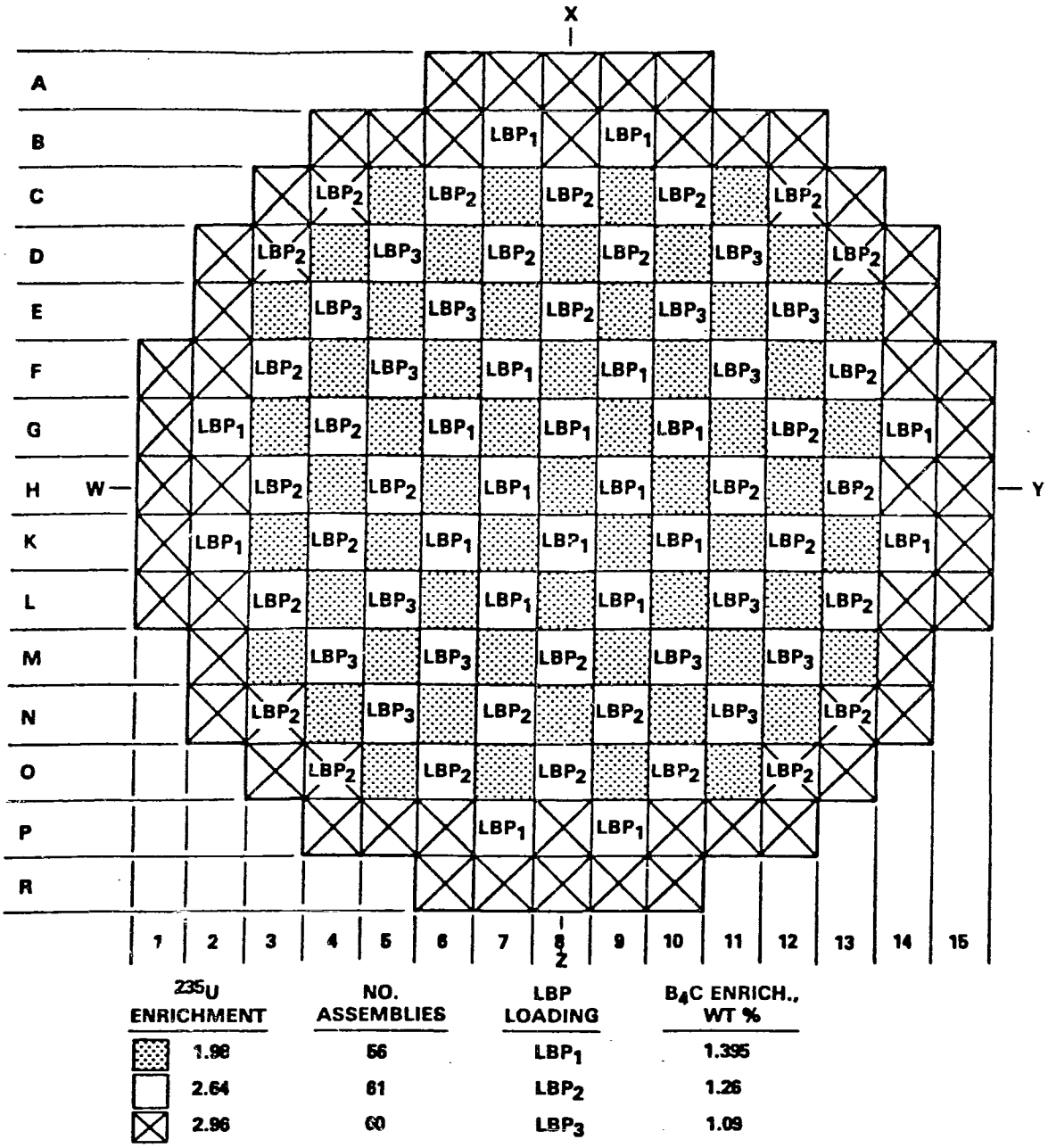


Figure C1. Location and Enrichment of Fuel Assemblies and Lumped Burnable Poison in TMI-2 Core.

SECTION D

TMI-2 FUEL ACCOUNTABILITY

The actual core damage will not be known until recovery operations begin. Therefore, accountability planning must be quite broad to cover a multitude of possible damage conditions.

TMI-2 accountability is described according to the following categories:

- Inventory difference and limits of error.
- Weighing of removed core components.
- Active and passive nondestructive assay (NDA) of breached or irregular fuel assemblies.
- Chemical verification of PWR code.
- Dispersed fuel and waste assay.
- SNM tracking system.

Figure D1 indicates methods for inventory of TMI-2 fuel.

An accountability task can be resolved into 2 components:

1. Bulk Measurement Accounting for the total SNM weight.
2. Specific Measurement Accounting for specific isotopes in expected proportions.

Neither of these measurements suffice alone. Bulk measurement may not indicate presence of specific fissile isotopes, and, conversely, specific measurement may not account for the bulk SNM quantity. Bulk measurement is a 100% category requiring a weight or some verification of every item. Specific measurement is sometimes carried out based on a statistical sampling plan in coordination with bulk measurements.

Accountability is essentially a bi-measurement task requiring verification by both shipper and receiver in any material transfer. Accordingly, diversion, or loss of material otherwise, can be sensed should a significant shipper-receiver difference be noted.

Since considerable fuel is expected to be recovered as irregular assemblies, and dispersed fuel forms, SNM certification of each recovered unit (fractional assembly, rubble container, etc) should be carried out for inventory verification and tracking. In the latter regard, a great number of samples will probably be sent to various laboratories, and SNM certification is essential to proper material control.

Weighing and NDA are described as bulk measurement techniques indicating the extent of core recovery, and therefore reflecting the aggregate of SNM removed from the core at any given time during recovery operations.

Chemical assay is described as a specific measurement technique serving primarily to substantiate (or calibrate if necessary) PWR computer code predictability. Thus, predicted quantities of uranium and plutonium isotopes can be relied upon for accountability purposes with a direct relation to chemical analysis based on a feasible sampling plan with respect to cost and core recovery efficiency.

Item accountability, provided for by 10CFR70.51 in normal reactor operations (see Appendix 1) for intact fuel assemblies entering or leaving a reactor, is considered a bulk measurement technique along with weighing and NDA for TMI-2 accountability. Item accountability may be quite limited for TMI-2, but should be applied to the extent that undamaged intact fuel assemblies are removed.

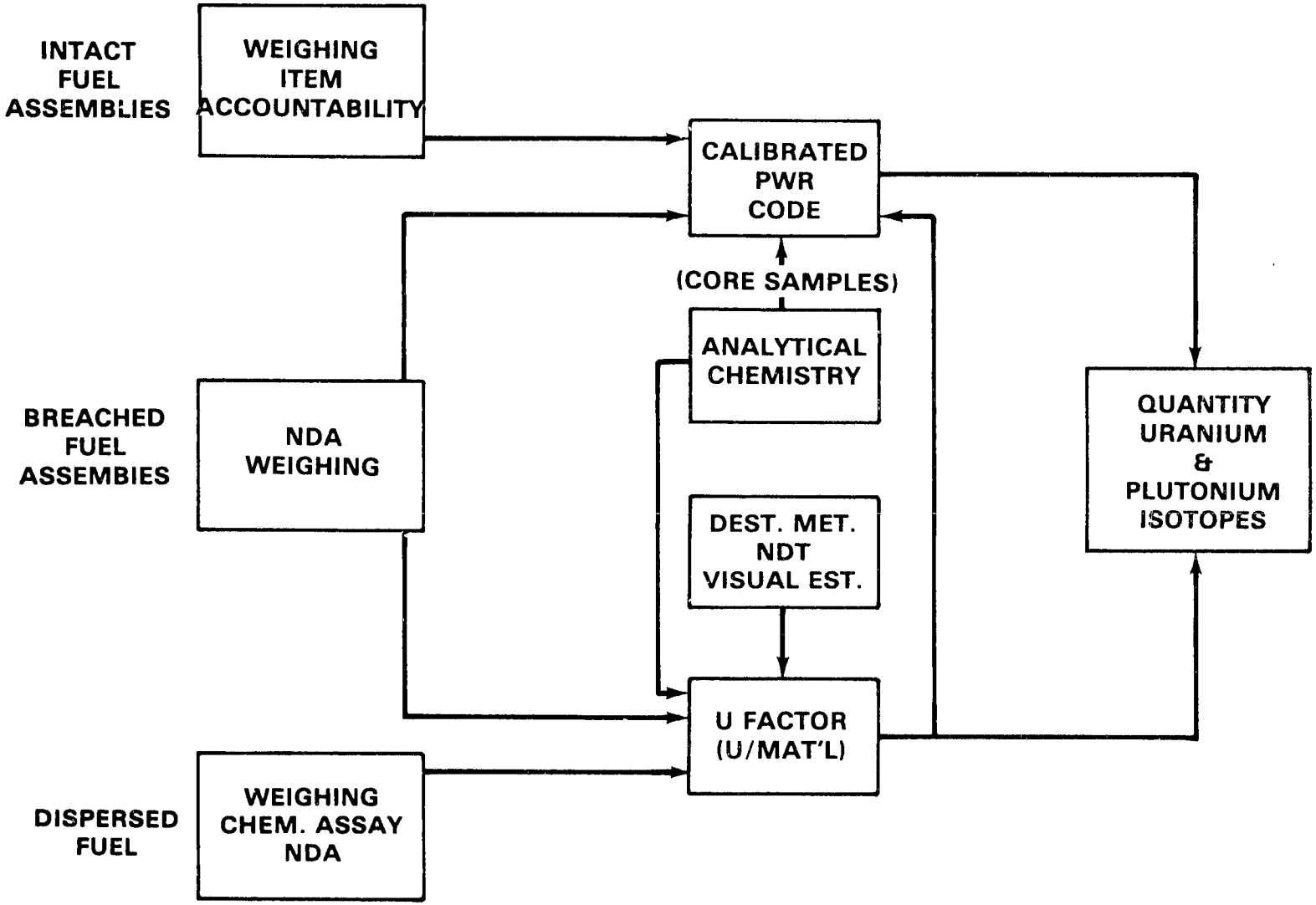


Figure D1. Methods for Inventory of TMI-2 Fuel.

INVENTORY DIFFERENCE AND LIMITS OF ERROR

TMI-2 operated for 94.6 effective full power days (EFPD) at the time of the accident. The average burnup has been estimated at 3176 megawatt days per metric ton heavy metal (MWD/MTHM) corresponding to approximately 0.321% of the total uranium or approximately 13.9% of the U-235 in the initial core loading. Burnup also includes fission related to production of Pu-239 and Pu-241.

Accountable SNM quantities are summarized as follows from values in Table C3.

	Initial Core Loading	Post-Accident (Predicted)
U-235	2,180 kg	1,914 kg
U-238	80,700 kg	80,490 kg
Total Uranium	82,900 kg	82,425 kg
Total Plutonium	---	159 kg

U-235 is depleted by fission and capture processes. The neutron capture-to-fission cross section ratio for U-235 in thermal reactors is 0.192. Essentially all of the U-235 capture product remains as U-236. U-238 is depleted by neutron capture forming Pu-239 by successive beta decays. The extent of U-235 depletion through fast fission is very small and considered negligible in this study. As Pu-239 builds up, depletion occurs through fission and capture processes. The thermal capture-to-fission cross section ratio for Pu-239 is 0.39. Successive radiative capture processes produce higher transuranic elements; however, plutonium depletion through these processes is considered negligible in this study. Some plutonium depletion will occur through buildup and fission of Pu-241. Burnup analysis takes this source of SNM depletion into account.

TMI-2 accountability should be concerned with balances on SNM quantities tabulated above, and possibly a balance on the total recovered core weight. Inventory difference and associated limits of error should be determined for

uranium quantities and the total core weight. Since plutonium was not present in the initial core loading, the post accident measured quantity of plutonium and its associated limit of error will be of concern.

Inventory Error Propagation

The general expression for inventory difference (ID) or Pu balance is given by:

$$Y = x_1 + x_2 + \ldots + x_n \qquad (D1)$$

The error for inventory difference of Pu balance can be approximated using the general expression:

$$(\sigma_y)^2 \cong \left(\frac{\partial y}{\partial x_1} \cdot \sigma_{x_1}\right)^2 + \ldots + \left(\frac{\partial y}{\partial x_n} \cdot \sigma_{x_n}\right)^2 \qquad (D2)$$

where:

y is the dependent variable (ID or Pu balance)

x represents independent variables (SNM quantities)

The inventory difference limit of error (LEID) is defined as $\pm$ two times the standard deviation of the total inventory difference measurement:

$$LEID \cong \pm 2\left[\left(\frac{\partial y}{\partial x_1} \cdot \sigma_{x_1}\right)^2 + \ldots + \left(\frac{\partial y}{\partial x_n} \cdot \sigma_{x_n}\right)^2\right]^{1/2} \qquad (D3)$$

(The same general approach applies to the limit of error for Pu balance.)

This expression is a reasonable error approximation if the standard deviations are small relative to the ID, and the covariance between the measured and estimated quantities is near zero.

Standard deviations in parentheses $\sigma_{x_1} \ldots \sigma_{x_n}$ are the geometric sums of the random and short term systematic errors which are expressed by:

$$\sigma_{x_n} \cong \left(\frac{\sigma_R^2}{N} + \sigma_S^2 \right)^{1/2} \qquad \text{(D4)}$$

Where σ_R is the random error in the measurement, N is the number of measurements and σ_S is distinguished from a bias (or long-term systematic error) in that bias has both a known magnitude and direction while σ_S has only a known magnitude.

Inventory Requirements

Tables D1, D2 and D3 provide inventory definitions, weight summaries, and inventory relations with associated limit of error expressions.

With respect to Table D2, the inset under (W) is typical, reflecting a measured weight and its associated variance. Weight entries are on a per item basis. Entries represent the assay method, as identified by Table D1, the observed weight, and the associated absolute variance for the measurement. Thus, w_i in Table D2 is the recovered weight of fuel assembly No. 1 (arbitrary) obtained by weighing (w) with an associated absolute variance of $(\sigma)^2$, where σ is the absolute standard deviation of the measurement.

Recovery totals at the bottom of Table D2 indicate summed values representing core totals along with corresponding summed absolute variances for item measurements. The recovery data are incorporated into final inventory and limit of error relationships of Table D3. (See Table D4 for a hypothetical LEID determination based on weighing.)

TABLE D1

TMI-2 INVENTORY DEFINITIONS*

BPu	Pu Burnup Wt.
BT	Total Burnup Wt. (U-235, Pu)
BU	U-235 Burnup Wt.
CR	Core Recovery
E	Estimated Wt. (Unrecoverable, or Long-Term Recoverable)
FT	Total Fissile Wt. (U-235, Pu)
I	Item Accountability
ID	Inventory Difference
LEID	Inventory Difference Limit of Error
NDA	Nondestructive Assay Wt.
NDAH	Process Holdup NDA Wt.
Negl.	Negligible
0	Initial Core Loading
P	Chemically Calibrated PWR Code Predicted Wt. per Fuel Assembly
PA	Post Accident
Pfr	P Applied to Fraction of Total U Wt. per Fuel Assembly
PuT	Total Pu Wt.
SCA	Sampling and Chemical Analysis
Uf	Uranium Factor - U Wt./Total Wt., by Chemical, NDT or Visual Est.
U5	U-235 Wt.
UT	Total Uranium Wt.
UTD	Total Uranium Wt. by Difference = [U Fac x (W_0 - W)]
Wi	Total Recovery Wt. per Intact Fuel Assembly by Weighing
Wb	Total Recovery Wt. per Breached Fuel Assembly by Weighing
Wo	Total Fuel Wt. per Fuel Assembly, Initial Loading
Wr	Total Recovery Wt. per Rubble Container (net) by Weighing
VCW	Cooling Water Volume Sampled
ΣBPu	Core Total Pu Burnup Wt.
ΣBT	Core Total Burnup Wt. [U-235, Pu(239, 241)]
ΣBU	Core U-235 Burnup Wt.
ΣFT	Core Total Fissile Wt. (U-235, Pu), Recovered
ΣU5	Core U-235 Wt. Recovered
ΣUT	Core Total U Wt. Recovered
ΣPuT	Core Total Pu Wt., Recovered
ΣW	Core Total Wt., Recovered, Including Non Fuel
$(\sigma)_i^2$	Absolute Variance Summation where i Refers to W, UT, etc. (See bottom row, Table D2)

*Wt. is per item basis. If preceded by Σ, Wt. is core total basis.

TABLE D-2

TMI-2 POST ACCIDENT INVENTORY WEIGHT SUMMARY

MATERIAL CATEGORY	(W) CORE RECOVERY WT.	(UT) TOTAL URANIUM WT.	(U5) U-236 WT.	(PuT) TOTAL PLUTONIUM WT.	(FT) (U-236, Pu) TOTAL FISSILE WT.	(BT) TOTAL (U-236, Pu) BURNUP WT.	(BU) U-236 BURNUP WT.	(BP) Pu BURNUP WT.
INTACT FUEL ASSEMBLY	1 $W_1, (\sigma)^2$ 2 : n	I P						
BREACHED FUEL ASSEMBLY	W_b	NDA SCA UTD Pfr						
DISPERSED FUEL								
RUBBLE	W_r	NDA SCA UTD	FOR SIGNIFICANT W_r (MAY BE NEGL.)					
COOLING WATER	SCA NDA VCW	FOR SIGNIFICANT SOLUTION TOTAL U (MAY BE NEGL.)						
PLATED OR LONG-TERM RECOVERABLE	NDAH	FOR SIGNIFICANT U (MAY BE NEGL.)						
WASTE	NDA SCA	FOR SIGNIFICANT U (MAY BE NEGL.)						
CORE RECOVERY TOTALS	$\overline{\Sigma W+\Sigma(\sigma)^2_W}$	$\overline{\Sigma UT+\Sigma(\sigma)^2_{UT}}$	$\overline{\Sigma U5+\Sigma(\sigma)^2_{U5}}$	$\overline{\Sigma PuT+\Sigma(\sigma)^2_{PuT}}$	$\overline{\Sigma FT+\Sigma(\sigma)^2_{FT}}$	$\overline{\Sigma BT+\Sigma(\sigma)^2_{BT}}$	$\overline{\Sigma BU+\Sigma(\sigma)^2_{BU}}$	$\overline{\Sigma BP+\Sigma(\sigma)^2_{BP}}$

REFERENCE TABLE D1

TABLE D3

TMI-2 INVENTORY RELATIONS AND LIMITS OF ERROR*

Inventory Difference	LEID	
Core Recovery = $\Sigma W_o - \Sigma W$	$2[\Sigma(\sigma)^2_{W_o} + \Sigma(\sigma)^2_W]^{1/2}$	(D5)
Uranium-Total = $\Sigma UT_o - (\Sigma UT + \Sigma BT + \Sigma PuT)$	$2[\Sigma(\sigma)^2_{UT_o} + \Sigma(\sigma)^2_{UT} + \Sigma(\sigma)^2_{BT} + \Sigma(\sigma)^2_{PuT}]^{1/2}$	(D6)
Uranium-235 = $\Sigma(U5)_o - (\Sigma U5 + \Sigma BU)$	$2[\Sigma(\sigma)^3_{U5_o} + \Sigma(\sigma)^2_{U5} + \Sigma BU]^{1/2}$	(D7)
Plutonium Balance	**LE**	
Pu Balance = Σ PuT	$2[\Sigma(\sigma)^2_{PuT}]^{1/2}$	(D8)

*Reference Tables D1 and D2.

WEIGHING OF REMOVED CORE COMPONENTS

A comparison of the known initial core weight and the recorded lift weight will provide a certain amount of accountability.

Lift weights would be an indication of recovered fuel at any time during recovery operations. When core recovery is complete, the total lift weight can be compared to the known initial core weight to indicate recovery efficiency. With essentially 100% recovery, including rubble, etc., from the pressure vessel and cooling system, the total lift weight would be expected to exceed the initial core weight slightly, depending on extent of zirconium cladding and UO_2 oxidation as well as moisture absorption and any corrosion effects on other core components.

Weighing will serve to estimate recovered SNM, but cannot stand alone as a measure of accountable material even though it may reflect complete recovery of fuel. Accountable species in the post accident total weight inventory will require quantitative verification by supporting measurements. PWR computer code predictions substantiated by chemical analysis could be utilized in support of the total weight inventory to quantify recovery of accountable U-235 and plutonium. Nondestructive assay can be utilized as a supporting measurement to weighing in evaluating SNM content of damaged fuel elements, rubble, etc as well. NDA is also described as a primary measurement method for total core SNM evaluation in this Section.

For breached fuel assemblies which have sufficient structural integrity to be lifted out of the water, it may be possible to use a tension load cell that is attached to the hoist cable to measure its weight. If the measurement is made on a dry basis, the fuel assembly will have to remain suspended long enough for the water to drain and evaporate off the hot surfaces of the fuel rods until the change in weight with time becomes negligible.

It may be necessary to weight the final assembly under water without exposing it to the air. This may be done, for example, by first weighing a perforated container holding the fuel assembly under water, and then evacuating it with pressurized nitrogen while the container is under water. The difference in weights will be the dry weight of the final assembly minus the buoyant force on the evacuated container. The error in making this measurement will be greater than obtaining the dry weight directly and the weighing will be more difficult to perform.

Weight as Primary Measurement

The mass measurement of uranium spent fuel is described as a primary technique for conducting a material balance. Table D4 statistically summarizes estimated data showing an LEID of approximately 632 kg, or approximately 0.76% of the initial loading. The expression for the inventory difference between the uranium weight of the initial core and damaged core after the accident is:

$$ID = W_0 - (W + W_E + BU + PuT + BPu)_{PA} \qquad (D9)$$

where W_0 is the initial weight of uranium in the core, W is the measured weight of uranium after the accident, W_E is the estimated weight of uranium in the fuel material (e.g., waste and unrecoverable material) which was not measured, BU is the U-235 burnup, PuT is the plutonium measured and BPu is the plutonium burnup. (D9) is the same as (D6) since BT = BU + BPu.
(D9) serves to identify all of the factors composing the total U ID.

The error in the ID can be approximated according to the general equation (D2, page D-5) as follows:

$$\sigma_{ID}^2 \approx \left[\frac{\partial(ID)}{\partial W_0} \cdot \sigma_{W_0}\right]^2 + \left[\frac{\partial(ID)}{\partial W} \cdot \sigma_W\right]^2 + \left[\frac{\partial(ID)}{\partial W_E} \cdot \sigma_{W_E}\right]^2$$

$$+\left[\frac{\partial(ID)}{\partial BU} \cdot \sigma_{BU}\right]^2 + \left[\frac{\partial(ID)}{\partial PuT} \cdot \sigma_{PuT}\right]^2 + \left[\frac{\partial(ID)}{\partial BPu} \cdot \sigma_{BPu}\right]^2 \qquad (D10)$$

The LEID is defined as two times the standard deviation of the total inventory measurement. The largest contribution to the LEID is from the measurement of the breached fuel rods which is discussed below and is shown in Table D4.

The "weight %" column in Table D4 is for the weight percent of uranium for each of the fuel categories. The number of samples, N_i, corresponds to the number of fuel assemblies for the intact and breached fuel and to the total number of fuel assemblies for the dispersed fuel. The objective is to segregate as much of the fuel material as possible by fuel assembly and not mix in one container material from more than one fuel assembly. w_i and W_i are the average sample weights (or average weight of fuel material in each container) and the total weight of the fuel material, respectively. The % Error is the uncertainty in estimating the weight of uranium in each container (or sample). It is an estimate which includes error contributions from sampling (probably the largest), weighing, and analyzing.

The quantity of uranium in breached fuel rods can be determined from the weight and U-factor by the following expression:

$$\text{Grams Uranium} = \text{Total Weight} \times \text{U-Factor}$$

The U-factor is the fraction of uranium in the fuel material and is affected by the degree of oxidation, water absorption, and separation of fuel from the cladding. It is estimated that up to 35% of the total Zircaloy in the core (where the total Zircaloy comprises 16% of the weight of a fuel assembly) has oxidized. The additional weight of oxygen would reduce the U-factor by approximately 1.5% from .7743. For badly damaged fuel assemblies, changes to the U-factor may also occur because of physical separation of the metal and fuel which may have occurred during the accident. For example, the fine fuel material (6% of the initial core inventory of fuel) which may have been carried by the coolant and deposited at other locations in the reactor may be expected to be very rich in uranium. This separation can cause another 5% absolute reduction in the U-factor giving a total bias

of (1.5 + 5) = 6.5%. Correction for this bias with an assumed uncertainty of one-half, or 3.25%, results in a short-term systematic error, σ_{S_i} = 3.25%.

The large systematic error in the uranium concentration can be eliminated or at least reduced by using analytical chemistry techniques (e.g., isotopic dilution or NDA of samples) on small samples of material removed from the fuel assembly. The isotopic composition can vary greatly with the location of samples as can be seen from the plots of burnup in the axial (Figure D2) and radial locations (Figure D3) in the core. The data if Figures D2 and D3 were taken from the PDQ fuel code runs shown in Appendix 3. (See also Figure C1 for U-235 variation.)

If the random error in weighing the breached fuel rods under water is 5%, then the LEID calculated in Table D4 is 632 kg uranium (or 0.8% of the total uranium inventory). The systematic error (σ_{S_i}) contributes 98 kg (or 15% of the total LEID) to the 632 kg LEID.

From these calculations it is seen that because of the large number of measurements which must be taken of the TMI-2 fuel, a small LEID may be achieved despite large measurement errors in each individual measurement. For this reason, an effective approach in reducing the Leid is in pursuing measurement techniques having smaller measurement errors by making actual measurements whenever possible, especially in reducing systematic errors. Since the measurement error for breached fuel rods will probably dominate the total inventory error, the effort to randomize the measurement errors should be directed at that category.

Weighing Techniques

Scales, balances and load cells are potential techniques for weighing the fuel material as it is extracted from the core. The use of scales and balances might require the transfer of the fuel material to a platform whereas a tension load cell may be able to be installed on the hoist so that

the fuel material is suspended by the hoist. Both load cells and scales are readily available with capacities exceeding the 660 kg required for weighing an entire PWR fuel assembly with an error of one part in 10^4 to 10^5. This is expected to be the largest mass that would have to be weighed for accountability purposes since this is the largest quantity which can be handled by the PWR spent fuel shipping cask.

Load Cells

Load cells are commonly used for loads ranging from 2,500 to 500,000 pounds (908 to 227,000 kg); however, rated capacities actually range from 1 to 10 million pounds (454,000 to 4.54 million kg). Standard load cells may be used at temperatures up to 115°F, but special load cells designed for higher temperatures are available.(6) Load cells are easier than scales to use in remote applications because an electrical lead of flexible hydraulic hose can link the platform to the console. Such a remote operation will be necessary for the core disassembly and removal because of the escape of large quantities of highly radioactive fission products into the pressure vessel. Because the strain gauge load cell produces an electrical signal proportional to the applied load, it readily interfaces with readouts, keyboards, computer terminals, and alarm systems.

Compression mounting for load cells is more frequently used than tension mounting. Compression load cells are normally used when the gross weight exceeds 10,000 lbs (4,526 kg). The universal load cell, which may be used in either compression or tension, is generally used in the compression mode between 1,000 and 10,000 pounds (454 to 4540 kg). In low capacity load cells, those used for 1,000 pounds or less, tension installations are more commonly used.

At the General Electric Company nuclear fuel fabrication facility in Wilmington, North Carolina, Orbitran and Toledo strain gauge load cells are used for measuring 20 to 45 kg cans of powder and 30B UF_6 cylinders.(7) Plans are currently being made to extend the application of load cells at

the Wilmington facility to the fuel rods weighing 3.5 kg and trucks. At the fuel fabrication facility operated by Combustion Engineering Corporation at Windsor, Connecticutt, applications for load cells are also being explored.(7)

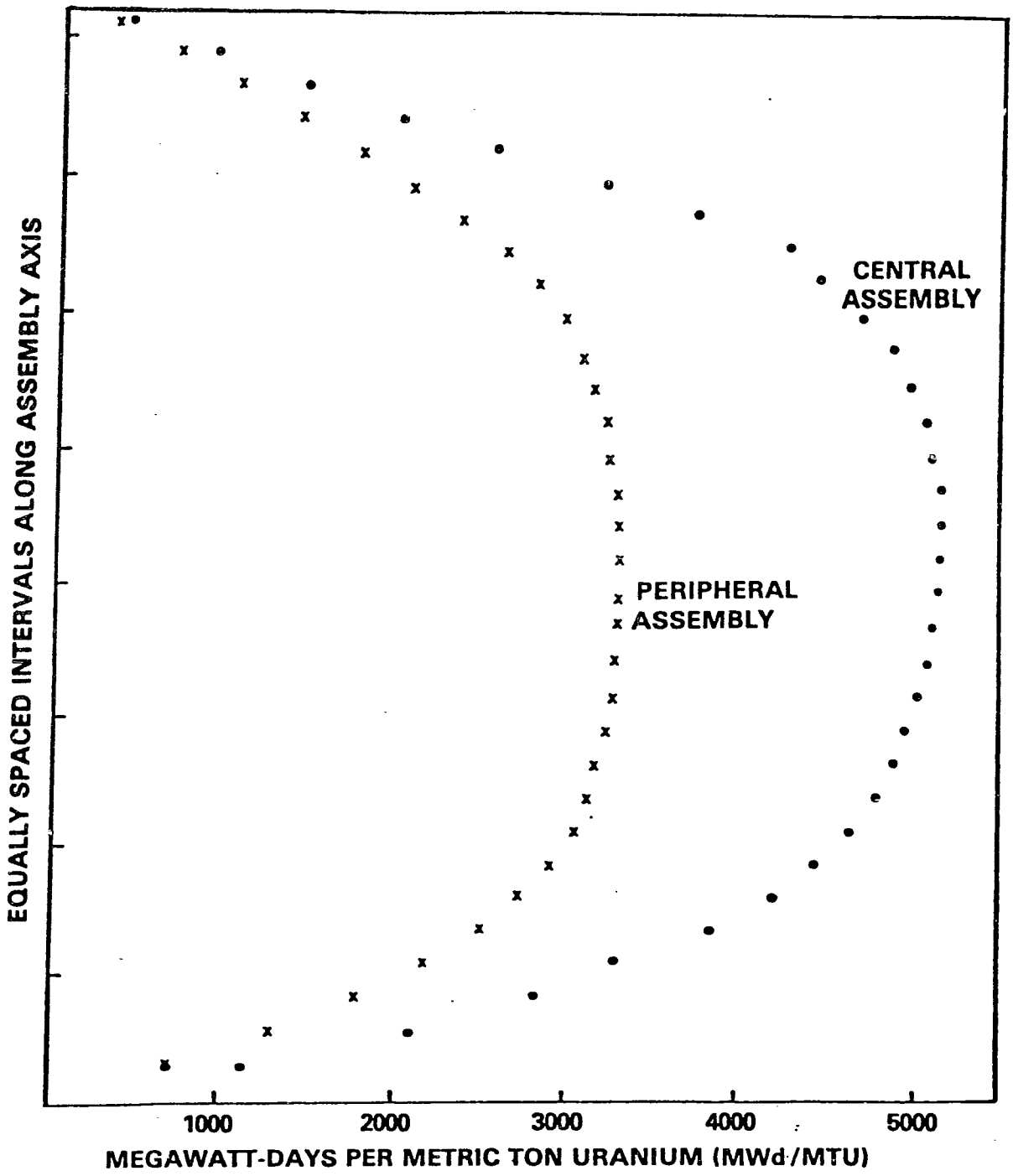


Figure D2. Axial Variation of Burnup in a Peripheral Fuel Assembly (H15 in Appendix 3) and in a Control Fuel Assembly (H8 in Appendix 3) in the TMI-2 Core. The vertical H-8 axis is arbitrary, equally spaced intervals.

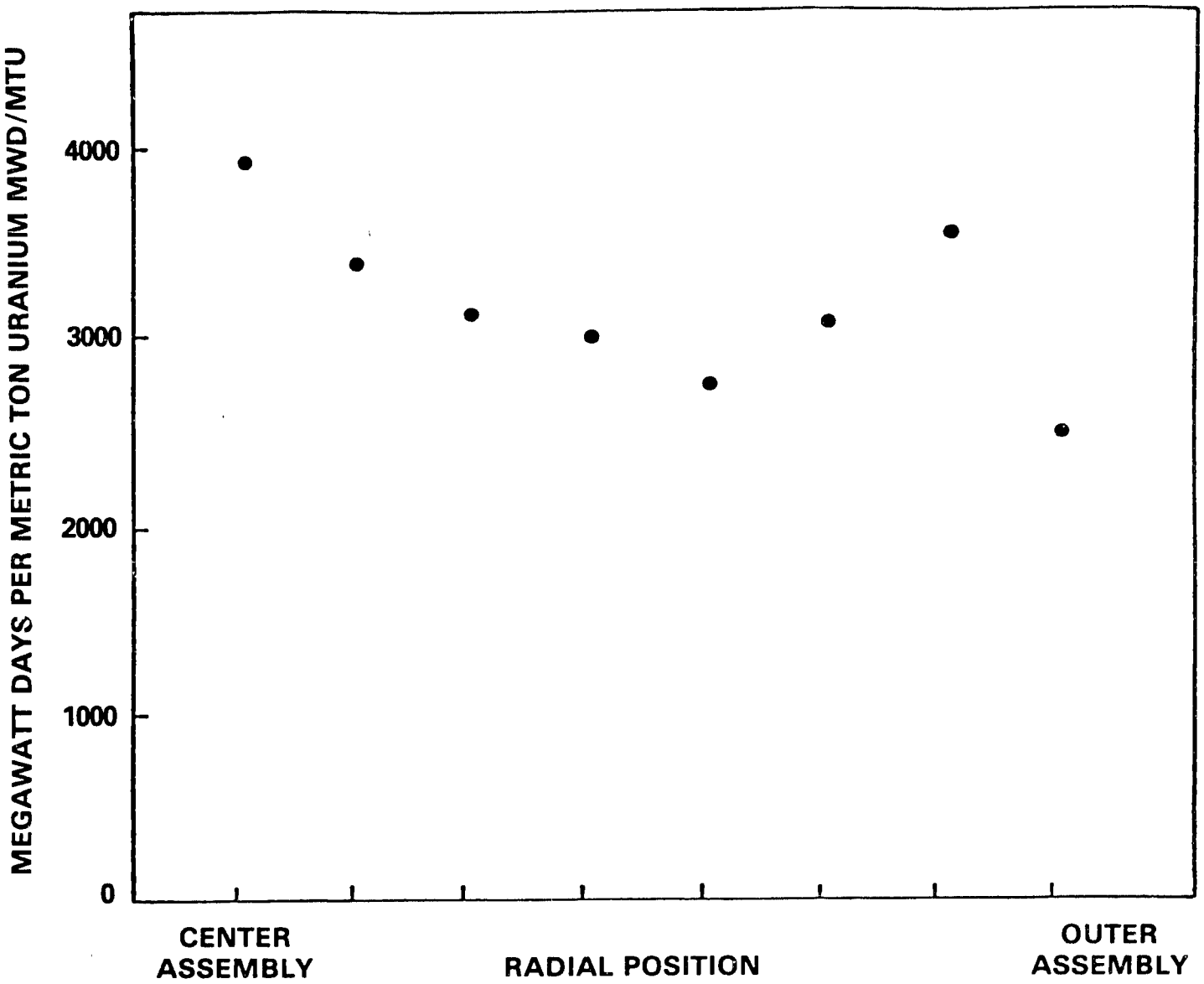


Figure D3. Radial Variation of Burnup in the TMI-2 Core.
Each point is the average burnup in a fuel assembly.

TABLE D4

HYPOTHETICAL LEID CALCULATIONS BASED ON WEIGHING TMI-2 BREACHED FUEL

Material Type	Weight %	N_i	w_i(kg)	W_i(kg)	% Error	σ_i(kg)	$N_i\sigma_i^2$(kg)^2	Fraction of Σ^2
Intact (Item Acc.)	23%	40	468.4	18,736	0	0	0	0
Breached	71%	137	427.7	58,595	5	22.00	6.631×10^{4}	0.927
Dispersed	6%	200	25	4,974	20[a]	5.00	5.000×10^{3}	0.070
Holdup/Plate	.003%	20	.15	3	50[a]	.08	1.280×10^{-1}	≈0
Waste	.001%	20	.05	1	30[a]	.02	8.000×10^{-3}	≈0
Test Samples	.056%	1000	.05	50	1	.00	0	≈0
Measured Plutonium	.186%	1	154	154	4.1[b]	6.31	3.982×10^{1}	≈0
Estimated Plutonium Burnup	.036%	1	30	30	4.0[b]	1.20	1.440	≈0
Estimated U-235 Burnup	.428%	1	355	355	4.0[b]	14.20	2.016×10^{2}	≈0
TOTAL				82,900			7.155×10^{4}	

$$\text{LEID} \cong \left\{\left[\sum_{i}^{i} N_i\,\sigma_i^2 + \Sigma_o^2 + N_{Bi}(\sigma_{Si}\,w_{Bi})^2 + N_{Di}\,(\sigma_{Si}\,W_{Di})^2\right] \times 4\right\}^{1/2}$$

where: $\Sigma_o = [N_i\ (0.1\% \times w_i)^2]^{1/2}$

$$\text{LEID} \cong \left\{[7.155 \times 10^4 + 6.2^2 + 137\ (.0325 \times 427.7)^2 + 200\ (.0325 \times 25)^2] \times 4\right\}^{1/2}$$

$= [177\ (.001 \times 468.4)^2]^{1/2}$

$\cong$ 632 kg U

$= 6.2$ kg

(a) Anticipated errors from imprecise NDA measurement
(b) From Reference 8

TABLE D4 (Continued)

DEFINITIONS OF TERMS

i = Material Type - Intact, Breached, Dispersed, Waste, etc

W_i = Total Weight of i

W = Total Weight of Inventory

N_i = Total Number of Samples

N_{Bi} = Number of Samples of Breached Fuel

N_{Di} = Number of Samples of Dispersed Fuel

w_o = Original Weight of Uranium per Fuel Assembly (sample)

w_i = Average Weight of Sample

W_{Bi} = Total Weight of Breached Fuel

W_{Di} = Total Weight of Dispersed Fuel

$$W = \sum_i^i W_i$$

$$W_i = N_i w_i$$

$\sigma_i \cong$ Average Absolute Error in W_i

$\sigma_{Si} \cong$ Relative Systematic Error in trying to correct for a bias created by a reduction in the concentration of uranium in the fuel

$\Sigma_i \cong$ Absolute Error in W_i

$\Sigma \cong$ Absolute Error in W

$$\Sigma_i^2 \cong \sum_i^{N_i} \sigma_i^2 \cong N_i \sigma_i^2$$

$$\Sigma_i \cong \sqrt{N_i}\ \sigma_i$$

$$\Sigma^2 \cong \sum_1^i N_i\ \sigma_i^2$$

$$W_o - W_i = ID$$

$$LEID^2 \cong (\Sigma^2 + \Sigma_o^2) \times 4 = (2\sigma)^2$$

$$LEID^2 \cong [\sum_1^i N_i\ \sigma_i^2 + \Sigma_o^2] \times 4$$

Wo = Original Core Loading Uranium Weight

$\Sigma_o^2 = \Sigma_{B\&W}^2$ Estimate of Original Inventory Error from Core Measurement at Babcock and Wilcox Company

NONDESTRUCTIVE ASSAY (NDA)

Nondestructive assay along with weighing may be considered a bulk measurement method for TMI-2 accountability. Passive and active techniques are described to evaluate fissile content directly in damaged fuel assemblies, rubble containers, etc. A third NDA technique is described which measures U-238 directly by means of 14 MeV interrogating neutrons. This method essentially reflects total uranium since U-238 comprises approximately 98% of TMI-2 fuel. The high energy neutron spectrum can then be tailored to provide thermal neutrons below the fission threshold of U-238, and a direct measure of U-235 and fissile plutonium is also possible. These techniques are described as follows:

Passive NDA with Fast Neutrons as Primary Measurement

The quantities of uranium-235 and plutonium in the recovered fuel can be measured by a passive neutron assay with fast neutrons. The measurement can be performed by determining the quantity of nuclides (e.g., plutonium and/or fissile uranium) in reference TMI-2 fuel assemblies which are used in establishing points on calibration curves relating the number of neutron counts to the grams of nuclide.

Fast neutrons are emitted by spontaneous fission of primarily plutonium-240 and by the (α,n) reaction primarily from plutonium-239 and -240. These neutrons are not significantly absorbed by the fuel materials or by the alumina-boron carbide (Al_2O_3 - B_4C) burnable poison and silver-indium-cadmium alloy poison which may be mixed with the fused and particulate fuel material. The neutron detector will therefore detect the fuel material in the middle as well as on the outside of the fuel assembly. To relate the number of neutron counts from the spent fuel assembly to the quantity of nuclide of interest (e.g., plutonium or fissile uranium), a calibration curve would have to be developed. This is difficult because there are three different enrichments (1.98 w/o, 2.64 w/o, and

2.96 2/o U-235) with which the fuel assemblies are loaded, and the burnup for each enrichment varies across the core. In addition, some of the fuel assemblies have burnable poison rods and some have silver indium-cadmium control rods.

The variation of neutron count rate with burnup (which is proportional to nuclide content within the range of burnup in the TMI-2 core) would have to be established for fuel assemblies having the same poison rod assemblies and the same enrichment. This would require five calibration curves. The variation of Pu-240 and U-235 with burnup for fuel having an original enrichment of 1.98 w/o U-235 is shown in Figure D4. The plots are from the tables in Appendix 3. The linear variation of nuclide concentration with burnup should also exist with the two other enrichments since the range of burnup is approximately the same for all enrichments.

There are several sources of error in performing a passive neutron assay which are discussed below. There is an uncertainty in determining the actual quantity of U-235 and plutonium in the "standard" fuel assemblies used in establishing the calibration curve. A comparison of the nuclide concentration in the dissolver solution of exposed LWR fuel measured chemically and that predicted by modified LEOPARD code was performed by the Yankee Atomic Electric Company.(8) The spent fuel had an initial enrichment of 4.94 w/o U-235 and a burnup varying between 17,729 and 32,030 MWD/MTUH. The concentration of nuclide predicted with the modified LEOPARD code differed from the measured values by 1 to 12% relative with the uncertainty for U-235 being 4% (A) (see equation D11). Results are summarized by Table D5.

Another uncertainty in establishing the calibration curve is in the number of counts for each "standard". This uncertainty is very small; however, because of the large number of neutrons emitted by the fuel assembly principally from Pu-239 and Pu-240. From the results of the ORIGEN calculations,(5) the total neutron emission rate [spontaneous + (α, n)]

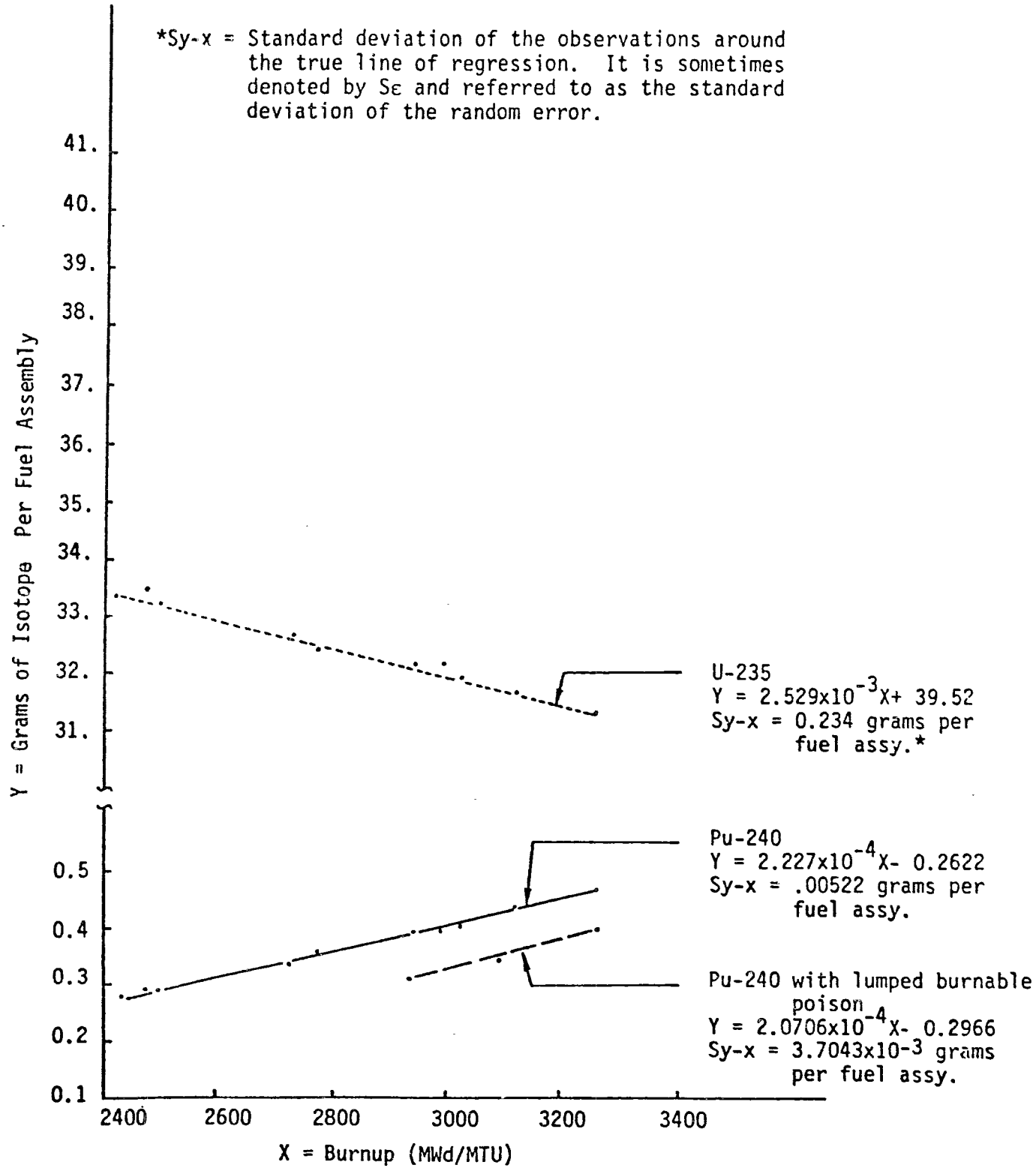


Figure 04. Variation of Grams of U-235, Pu-240 with Burnup

TABLE D5

COMPARISON OF NUCLIDE QUANTITIES PREDICTED BY MODIFIED LEOPARD CODE AND MEASURED QUANTITIES SAMPLED FROM DISSOLVER SOLUTION OF EXPOSED LWR FUEL (8)

Nuclide	Difference*
U-235	Within 4%
U-235	Within 4.5%
Pu-239	Within 4.5%
Pu-240	Within 1.5%
Pu-241	Within 3.1%
Pu-242	Within 12%
Fissile Pu	Within 4.1%
Total Pu	Within 3.5%
Total U	Within 1.0%

*$\%\ \text{Difference} = \frac{\text{Measured - Calculated}}{\text{Measured}} \times 100$

for the TMI-2 core 1.5 years after shutdown is estimated to be 2.198×10^7 n/s-core. For 177 fuel assemblies, the average emission rate for a fuel assembly is 1.2×10^5 n/s-fuel assembly.

For neutron detectors with 5% intrinsic efficiency, and a detector system intercepting one-half of the neutrons emitted from the fuel assembly source due to loss of neutrons traveling outside the solid angle created by the source and detector surface, the number of neutron counts is $.04 \times .5 \times 1.2 \times 10^5 = 2.1 \times 10^3$ counts/second-fuel assembly. The number of counts over a one-minute counting period in which the fuel assembly is held in position for one minute before being moved axially by an increment of roughly the height of the detector ring (2 to 3 feet) is $3.1 \times 10^3 \times 60 = 1.86 \times 10^5$ counts/fuel assembly. The standard deviation as a percent of the mean for this number of counts is approximately $\frac{1.9 \times 10^5}{1.9 \times 10^5} = 0.23\%$ (B), (see equation D11).

The uncertainty in the measurement of a fuel assembly caused by the transfer of material of one enrichment to a fuel assembly of another is estimated at ~2% (C), (see equation D12). The rationale for this estimate is that only material with particle size less than 0.06 cm (activated at 6% of the core) could be suspended by the coolant and transported. Only a fraction of this would have been deposited in the core with the remainder being carried by the coolant into other areas of the coolant system.

The uncertainty for counting statistics in measuring the recovered fuel assembly should be the same as for the standards, namely 0.23%. Two types of errors can be calculated--the systematic error or uncertainty in the calibration curve (SE_{CAL}), and the random error (RE_{MEA}). The uncertainty in measuring the recovered fuel due to differences in geometry (bulk density of the fuel material and source-detector distance) between reference TMI-2 fuel calibration assemblies and recoverd fuel assemblies is treated as a component of RE_{MEA} and estimated at 10% (D) (see equation D12). Effects of B_4C and silver-indium-cadmium control rod materials

inhomogeneously mixed with the fuel contribute an estimated uncertainty of ~1% (E), (see equation D12), when the measurement is carried out in the dry condition.

Values for calibration and measurement errors can be approximated as indicated by (D11) and (D12) below. Sample weights (w_i) and total weights ($_i$) in Tables D4 and D6 are essentially the same even though they are for total uranium and U-238 respectively. U-238 is approximately 98% of the total uranium, and the difference in the LEID will be ~20 kg. The fraction $\left(\frac{\text{LEID}}{\text{Total Inventory}}\right)$ will remain about the same on a total uranium or U-238 basis.

$$SE_{CAL} \cong \sqrt{A^2 + B^2} \cong \sqrt{.04^2 + .002^2} \qquad \text{(D11)}$$

$$\cong .04$$

$$RE_{MEA} \cong \sqrt{D^2 + E^2 + C^2 + B^2} \cong \sqrt{0.1^2 + .01^2 + .02^2 + .002^2} \qquad \text{(D12)}$$

$$\cong 0.10$$

The LEID based on measuring breached fuel assemblies by passive NDA ($RE_{MEA} \cong 10\%$) is 1119 kg. This calculation is given by Table D6. Note by Table D4 that the LEID is 632 kg when breached fuel is measured by weighing ($RE_{MEA} \cong 5\%$).

Active NDA as Primary Measurement

The use of sub-MeV neutrons from sources such as Sb-124, Be, with intensity of 6×10^6 neutrons per curie of Sb-124, and moderated Cf-252 (2.34×10^{12} n/sec/gram) induce thermal fission in U-235, Pu-239, Pu-241 without significant fissioning of the abundant U-238. The degree of neutron spectrum softening in each of these sources (Be mantel for the Sb-Be and Be with polyethylene for the Cf-252) is dictated by the required signature

TABLE D6

HYPOTHETICAL LEID CALCULATIONS BASED ON PASSIVE NEUTRON COUNTING OF TMI-2 BREACHED FUEL*

Material Type	Weight %	N_i	w_i(kg)	W_i(kg)	% Error	σ_i(kg)	$N_i\sigma_i^2$(kg)^2	Fraction of Σ^2
Intact (Item Acc.)	23%	40	468.4	18,736	0	0	0	0
Breached	71%	137	427.7	58,595	10	44.0	2.652×10^{5}	0.981
Dispersed	6%	200	25	4,974	20[a]	5.00	5.000×10^{3}	0.018
Holdup/Plate	.003%	20	.15	3	50[a]	.08	1.280×10^{-1}	≈0
Waste	.001%	20	.05	1	30[a]	.02	8.000×10^{-3}	≈0
Test Samples	.056%	1000	.05	50	1	.00	0	≈0
Measured Plutonium	.186%	1	154	154	4.1[b]	6.31	3.982×10^{1}	≈0
Estimated Plutonium Burnup	.036%	1	30	30	4.0[b]	1.20	1.440	≈0
Estimated U-235 Burnup	.428%	1	355	355	4.0[b]	14.20	2.016×10^{2}	≈0
TOTAL				82,900			2.704×10^{5}	

$$\text{LEID} \cong \left\{\left[\sum_{1}^{i} N_i\,\sigma_i^2 + \Sigma_o^2 + (N_j\,(SE_{CAL} \times w_i)^2\right] \times 4\right\}^{1/2} \qquad \text{where: } \Sigma_o = [N_i\,(0.1\% \times w_i)^2]^{1/2}$$

$$\text{LEID} \cong \left\{[2.704 \times 10^5 + 6.2^2 + 137\,(.04 \times 427.7)^2] \times 4\right\}^{1/2} \qquad = [177\,(.001 \times 468.4)^2]^{1/2}$$

$$\cong 1119 \text{ kg U} \qquad = 6.2 \text{ kg}$$

(a) Anticipated errors from imprecise NDA measurement
(b) From Reference 8

*See Table D4 for definitions

intensity of prompt fission neutrons. The intensity increases with spectrum softeners and self shielding. The signature intensity should be high enough to minimize background effects from neutrons emitted by Pu-240 which provide the basis for passive neutron measurements.

In order to distinguish between U-235 and Pu-239, the neutron spectrum is tailored for separate observations to determine the isotopic content of each fissile isotope. Based on Babcock & Wilcox (B&W) in-house calculations, the quantity of Pu-239 in the TMI-2 core at the time of shutdown varies from 11.3% of the U-235 in Region 1--containing 1.68% U-235, to 5.0% of the U-235 in Region 3--containing 2.66% U-235. The core average Pu-239 concentration is 7.7% of the U-235 present. Because of the small amount of Pu-239 in the TMI-2 fuel, the two-spectra type measurement will require long counting periods and very careful measurements for reasonable accuracy. These conditions might limit the technique to spot checks and confirmatory measurements. The technique would be very effective in determining the fissile (mainly U-235) to fertile (U-238) ratio, however.

In order to determine the fissile to fertile ratio, neutron spectra above and below the fission threshold of U-238 are required. The induced fissions are measured via the delayed neutrons:

$$Y_1 = \alpha_{18} M_{28} + \alpha_{15} M_{25} \qquad \text{Sub Threshold Spectrum} \qquad \text{(D13)}$$

$$Y_2 = \alpha_{28} m_{28} + \alpha_{25} M_{25} \qquad \text{Above-Threshold Spectrum} \qquad \text{(D14)}$$

Where Y_1 represents delayed neutron yields for the two-neutron spectra; α_{18} and α_{15} are the effective numbers of delayed neutrons per unit mass of U-238 and U-235 respectively, in the case of the sub-threshold spectrum $\alpha_{15} > \alpha_{18}$, and in the case of above-threshold spectrum $\alpha_{28} \gg \alpha_{25}$. Calibration assemblies to determine α_{18} and α_{15}, can be provided by fresh fuel assemblies with three different enrichments and a natural uranium assembly. An inherent disadvantage in this technique is the relatively high background relative to the weak delayed neutron signature from (α,n) and spontaneous fission neutrons in Pu-241.

Calibration Errors (SE_{CAL})

- Uncertainty in quantity of U-235 in the reference standards $\sigma \cong 1.0\%$
- Uncertainty in the number of counts* $\sigma \cong 0.1\%$

Measuring Errors (RE_{MEA})

- Uncertainty from the difference in geometry and selfshielding of the fuel in the TMI-2 fuel assemblies $\sigma \cong 10\%$
- Uncertainty in the number of counts $\sigma \cong 0.1\%$

The total error in calibration is

$$SE_{CAL} \cong \sqrt{.01^2 + .001^2} \qquad \text{(D15)}$$
$$\cong .01$$

The total error in measurement is

$$RE_{MEA} \cong \sqrt{0.1^2 = .001^2} \qquad \text{(D16)}$$
$$\cong 0.10$$

When the 10% measurement error for breached fuel is expressed as U-238 (Table D7), the LEID is 1119 kg U-238 which is the same as for the passive measurement.

A cross-section of a conceptual active NDA system for measuring TMI-2 fuel assemblies is shown in Figure D5 (Ref. 9). Lead shielding around the fuel assembly reduces gamma radiation to levels required for various detectors. Fission chambers and B-10 coated proportional counters need little, if any, lead shielding, and can operate near or even inside the fuel assembly. BF_3, He-3, He-4, or proton recoil detectors--liquid, solid, or gaseous--

*This assumes a Cf-252 source on the order of 2.5×10^{10} neutrons/second, a detector efficiency of 0.28 and a delayed neutron signal of 1.4×10^{-4} cps per neutron from a Cf-252 source (see Ref. 9).

will not sustain the high gamma fields, and will require massive lead shielding. A 17.78 cm thick lead shield surrounding a 19.71 x 19.71 cm^2 square fuel assembly similar to the one used in the San-Onofre Pressurized Water Reactor can be used in this system. The amount of detector shielding for TMI-2 fuel will be substantially smaller because of the lower burnup. The main lead shield is cylindrical and may be supplemented by lead sectors added to the fuel assembly (see Figure D5). The lead shielding is surrounded by a thin (1.27 cm thick) polyethylene (CH_2) moderator. The principal function of the moderator is to further slow down prompt and delayed neutrons originating in the fuel assembly. Surrounding the CH_2 moderator is a ring of detectors (5.8 cm in diameter, 90 mm Hg pressure) with 2.54 cm thick CH_2 flat spacers interposed between them. Behind the detector ring, there is a 7.62 cm thick layer of CH_2 as a thermal neutron reflector.

Active NDA Systems Using 14 MeV Neutron Sources

High intensity beams of 14 MeV neutrons can be produced inexpensively by small, low voltage (typically 150 kV) accelerators, such as the Cockcroft-Walton Accelerator. The reactor that serves to produce the neutrons is D (T, He-4) n + 17 MeV.

The advantage of an accelerator, rather than a neutron source, is that the accelerator can be turned off, eliminating the interrogating neutron background. This is particularly important when measuring delayed neutrons with pulsed penetrating neutron sources. If two interrogations are made of the TMI-2 fuel, one with a beam of neutrons having energies mostly above the fission threshold for U-238, and the other with a tailored spectrum of neutron energies below the fission threshold, the isotopic concentration of U-235 and U-238 can be measured. The measurement principles are similar to those discussed in the previous section.

TABLE D7

HYPOTHETICAL LEID CALCULATIONS FOR ACTIVE NEUTRON COUNTING OF TMI-2 BREACHED FUEL*

Material Type	Weight %	N_i	w_i(kg)	W_i(kg)	% Error	σ_i(kg)	$N_i\sigma_i^2$(kg)^2	Fraction of Σ^2
Intact (Item Acc.)	23%	40	468.5	18,736	0	0	0	0
Breached	71%	137	427.7	58,595	10	44.0	2.652×10^{5}	0.981
Dispersed	6%	200	25	4,974	20(a)	5.00	5.000×10^{3}	0.018
Holdup/Plate	.003%	20	.15	3	50(a)	.08	1.280×10^{-1}	≈0
Waste	.001%	20	.05	1	30(a)	.02	8.000×10^{-3}	≈0
Test Samples	.056%	1000	.05	50	1	.00	0	≈0
Measured Plutonium	.186%	1	154	154	4.1(b)	6.31	3.982×10^{1}	≈0
Estimated Plutonium Burnup	.036%	1	30	30	4.0(b)	1.20	1.440	≈0
Estimated U-235 Burnup	.428%	1	355	355	4.0(b)	14.20	2.016×10^{2}	≈0
TOTAL				82,900			2.704×10^{5}	

$$LEID \cong \left\{\left[\sum_{1}^{i} N_i\,\sigma_i^2 + \sum_{o}^{2} + (N_i\,(SE_{cal} + w_i)^2)\right] \times 4\right\}^{1/2}$$

$$LEID \cong \left\{[2.704 \times 10^5 + 6.2^2 + 137\,(.04 \times 440)^2] \times 4\right\}^{1/2}$$

$$\cong 1119 \text{ kg U}$$

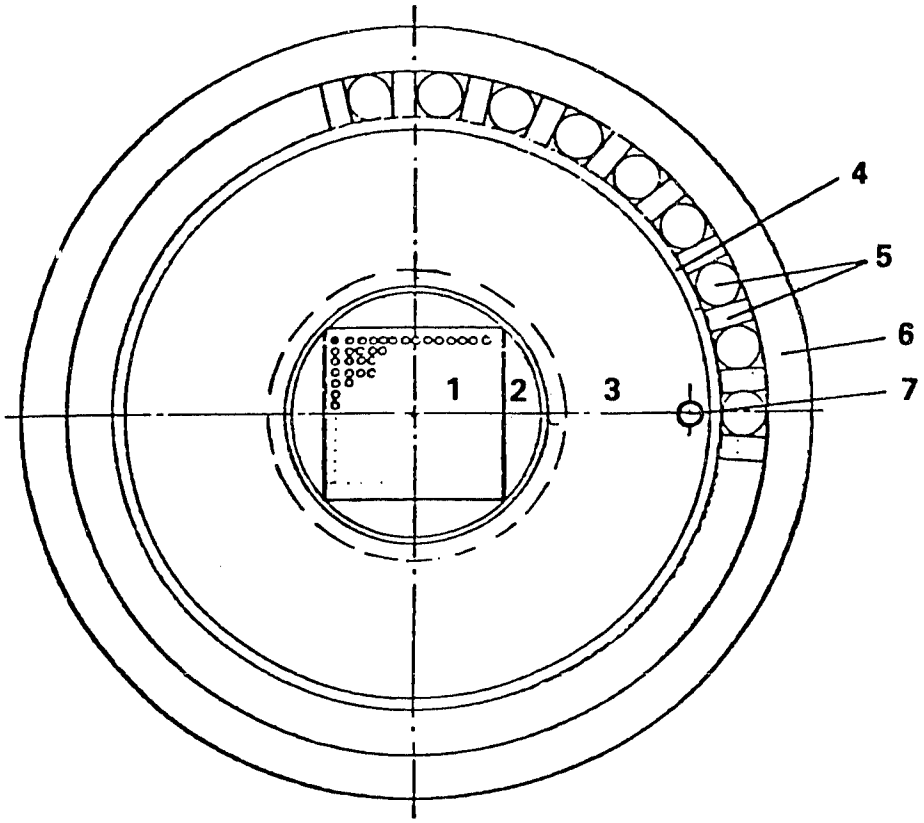
where: $\Sigma_o = [N_i\,(0.1\% \times w_i)^2]^{1/2}$

$= [177\,(.001 \times 475)^2]^{1/2}$

$= 6.32$ g

(a) Anticipated errors from imprecise NDA measurement
(b) From Reference 8

*See Table D4 for definitions



1. PWR FUEL ASSEMBLY CONSISTING ABOUT 200 LOW-ENRICHED URANIUM FUEL PINS
2. LEAD SEGMENTS
3. LEAD SHIELDING
4. POLYETHYLENE (CH_2) MODERATOR
5. BF_3 TUBES AND POLYETHYLENE SPACERS
6. POLYETHYLENE (CH_2) REFLECTOR
7. POSSIBLE SOURCE LOCATION

Figure D5. Cross Section of a Generic Measuring System for Active Neutron Counting.

Because 14 MeV neutrons can penetrate well in most materials, and especially in hydrogeneous substances, the 14 MeV delayed neutron method is particularly suited for measuring the content of large volume containers from 1 gallon cans to 55 gallon drums containing scrap and waste materials. Lead is virtually transparent to these neutrons, and they can also be used in measuring uranium and plutonium content of spent fuel elements in thick lead casks. Most delayed neutron detectors are based on thermalization, by a hydrogeneous moderator, of the delayed neutrons and then counting the thermal neutrons in BF_3 or ^3He counters. These detector systems are fairly insensitive to gamma rays, hence the high gamma background from samples will not affect the measurement when the detectors are well shielded. The technique is also applicable to measuring fissile material in waste which is contaminated with fission products.

NDA General Considerations

All NDA techniques are affected by the geometry of the fissile material. Assays with accuracy of a few percent require calibration with standards physically as similar to the unknown as possible. This will be impossible for TMI-2 because of the anticipated variations in breached assemblies. Rods will have varying lengths. Fractured fuel pellets and ZR-UO_2 eutectic material may be piled on the stubs of rods and the inconel spacers. Thus, the linear density of fuel may vary from zero to several times the original linear density. Some enrichment mixing may have occurred from pellets mixing with those from adjacent assemblies. A complete assay system with accuracy comparable to that achieved with fresh fuel is beyond the capabilities of current technology and may not be required at TMI-2. The challenges of TMI-2 will do much to evaluate and improve spent fuel assay techniques.

NDA of spent fuel can be based on gamma or neutron techniques. Gamma spectroscopy measures the activities of fission products which are proportional to burnup. The main uncertainties arise from gamma attenuation in the fuel and fission product migration. With breached fuel, both problems are

aggravated. Neutron methods measure the neutrons emitted by the fuel due to spontaneous fission and (α,n) reactions with light elements. This can be correlated with burnup. Active techniques measure fissile content by exposing assemblies to a radioisotope or accelerator produced neutron flux and measuring the induced prompt or delayed neutrons.

Summary of NDA Techniques

Spent fuel assay techniques have been reviewed by Hsue, et al.[10] Methods based on reactivity, calorimetry, neutron and gamma measurement are discussed there and summarized here.

Reactivity measurements have been used for many years to test new fuel elements. This technique could be expanded to complete or partial assemblies. A hardened flux would be required to minimize the variation of self-shielding caused by the core damage. Such an assay system would be complex and costly, requiring a precise critical or sub-critical assembly. Much development would be required in addition to detailed calculations but this measurement could possibly be applied to all fuel types.

Calorimetry of spent fuel measures the heat generated by fission products and transuranic isotopes. Typically, this is a slow measurement due to the time required for thermal equilibrium. This time will be geometry dependent. Accuracies of a few percent have been achieved for small (liter size) samples. For spent fuel assay the isotopics will have to be calculated or measured by another technique and input along with fission product inventory calculations. These are the same inputs required for passive neutron or gamma measurement interpretation. In addition, calculations need to be performed to determine the fraction of gamma flux absorbed in the container. Calorimetry can be carried out on sealed containers, but the container thermal mass further slows the measurement. Calorimetry of extended fuel assemblies would need further development before it could be applied to TMI-2.

Gamma spectroscopy measures fission product activity. This can be correlated with burnup for fixed geometry assemblies, otherwise, it is proportional to the amount of fuel present and its burnup. Activity ratio measurement has been developed that requires measurement of a fission product and an isotope formed by (n,α) on a fission product. The ratio of the latter to the former will still be burnup dependent (proportional to integrated neutron flux). But because it is a ratio measurement absolute detection efficiencies are not required.

Passive neutron measurements have the advantage that neutrons are less attenuated in assemblies than gamma rays and, therefore, uncertainties due to geometry and self-shielding are reduced. Irradiated fuel with normal end-of-cycle burnups emit 10^7 to 10^8 n/s per assembly. ORIGEN code data predicts an average of 1.2×10^5 n/s from low burnup TMI fuel. The (α,n) and spontaneous fission sources are roughly equal and arise primarily from the even plutonium isotopes.

Active neutron techniques have the attractive feature of measuring directly fissile content. The sample to be assayed is exposed to epithermal neutrons and the induced fission rate is monitored by measurement of prompt or delayed neutron emission. There are roughly 200 times more prompt than delayed neutrons. Delayed gamma emission has been used for unirradiated fuel. The irradiating neutrons can be produced by Cf-252, an antimony-beryllium (Sb-Be) source, or an accelerator. For active techniques, the passive neutrons are an interfering background. In water, multiplication and self-shielding will be important.

An NDA technique to be applied to TMI-2 should consider the following:

- Relatively independent of fuel geometry
- Low maintenance/remote operation
- Easily interfaced to a material access and control computer system
- Accuracy
- Compatibility with core disassembly procedure

Of the techniques discussed above, fission product gamma scanning, passive neutron scanning, and active neutron assay offer some attractive features which are discussed below:

Weight measurement is a gross indication of fissile content, but has the following problems. Not all material may be weighable, e.g., plate-out and holdup. The U-weight fraction of the total weight will vary due to separation of pellets from the cladding and oxidation. Weight, even for intact assemblies, is not a direct measure of fissile material, but must be supplemented by isotope dilution/mass spectroscopy, burnup calculations to determine fissile content, or NDA. Large sampling errors are likely to occur if reliance is placed on isotopic dilution. (See limited sampling plan under chemical verification of PWR code in following section.) Neutron or gamma scanning can distinguish between fuel and other core materials, can measure holdup or plate-out, and provides a more direct measure of fissile material. With an active assay, fissile weight is measured directly.

Problems with Applying NDA

The variable geometry and self-attenuation of the fuel lead to uncertainty in the NDA result. The passive neutron measurement in air should be more accurate than gamma assay. Neutrons are relatively unaffected by self-shielding and thus accurately measure interiors of assemblies. This is not true, however, if the measurement is made in borated water, since neutrons are rapidly thermalized and absorbed and the effect of geometry and self-shielding increases. A measurement in air complicates the core removal process.

The amount of gamma ray self-absorption is energy dependent. The 0.66 MeV gamma line of Cs-137 is 39% self-absorbed in a rod, whereas the 2.18 MeV gamma line, from Ce-144, is only attenuated by 15%. The effect is more pronounced for interior rods in an assembly. Only 1% of the 0.66 MeV activity from an interior rod is transmitted, about 40% of the 2.18 MeV line is transmitted.

If fuel is missing from outer rods then gross Ce-144 activity will be overestimated. The passive neutron emission worth of rods in bundle only varies by 20% for thermal flux and 10% for fast flux. Neutron multiplication is, however, 50% for intact bundles [11] and this is sensitive to geometry.

The fission products monitored should have long half lives, low migration in the fuel, and easily resolved intense high energy gamma lines. Cs-137 and Cs-134/Cs-137 ratios have been studied most extensively. However, cesium migrates out of fuel pellets and this will be very pronounced for breached fuel and thus not applicable to TMI-2.

Eu-154 [an isotope produced by (n,α) or a fission product] and Ce-144 do not migrate and their ratio could be a good burnup monitor. Absolute Ce-144 activity would be proportional to the fuel present times the neutron specific count rate. Ce-144 has a 2.18 MeV gamma line that is clearly resolved in fuel with cooling times of over a year. But even for such a high energy gamma ray, self-shielding and shielding of the inner rods of a bundle will be significant. Calculations must be performed to correct for shielding.

The shielding is strongly dependent on geometry. With damaged assemblies, such calculations may not be possible. Then a transmission measurement or self-indication of attenuation using differential absorption techniques will be required. The differential absorption technique monitors the intensity ratio of two gamma lines from the same isotope.

Besides these problems with measurement accuracy, there is further uncertainty introduced in deducing the fissile mass from the NDA measurement. The neutron or gamma count rates per gram of fuel are proportional to the number of fissions that have occurred in the fuel. This in turn depends on the core location, neutron spectrum, and original enrichment, all of which affect burnup. Burnup calculations, destructive assay of fuel samples, and measurements on intact assemblies must be used to

derive a factor to relate the counts from a given assembly or subassembly to grams U-235. Due to the large variations in burnup, this factor, a strong function of core location, will be difficult to establish accurately.

Considering these effects, relatively intact bundles could have a measurement error of ± 10% with passive neutron or gamma counting. For very damaged fuel, the measurement errors could be ± 20% by neutron counting (corrected for multiplication) and ± 30% by gamma counting (if a density correction is made). Making neutron or gamma counting measurements requires the preparation of calibration samples. The concentration of isotopes in these samples must be determined either by an analytical technique such as isotopic dillution on grab samples or by the results of a burnup code (e.g., ORIGEN, PDQ, or LOR). The errors in sampling or in the burnup code (which ranges from 5% to 10%) will contribute significantly to the absolute total measurement error.

The supplemental active assay by a shuffler, for example, faces similar difficulties; however, the source flux can be hardened and a bulk density correction can be made for multiplication giving an estimated accuracy of ± 15%.

For holdup of fuel in the reactor coolant systems and plate-out of uranium dissolved in the coolant, fission product gamma ray measurement may be the only applicable technique. Accuracies of 30 to 50% are likely. The major uncertainty being solid angle and attenuation calculations.

CHEMICAL VERIFICATION OF PWR CODES

PWR computer code predictions provide a means to evaluate urnaium and plutonium isotope quantities for given burnup levels. Individual fuel assembly values are predicted and can be summed to give core total values. (See Appendix 3.) Sampling and chemical analysis of each fuel assembly would undoubtedly provide a much more reliable means to evaluate total core SNM values. However, the number of samples required may impact recovery operations, and laboratory costs would certainly be a significant factor. Consequently, a limited sampling and chemical assay program is proposed to substantiate (and calibrate if necessary) PWR code predictability. Accordingly, code values would be directly related to a chemical primary standard so to speak and applied as necessary to a multitude of assay requirements in TMI-2 accountability.

Sampling would be carried out over a number of selected fuel assemblies covering the burnup range. A comparison of predicted values with chemical results would reflect code predictability, and establish any consistent differences or bias. In the latter event, it may be possible to make adjustments to particular codes in use and bring results into agreement with chemical values.

Core sampling and chemical analysis were carried out on JPDR-1 spent fuel.[12] Both radiochemical and chemical assay techniques were applied. Chemical data were based on the triple spike isotope dilution technique described below. TMI-2 core sampling and essay will have interesting parallels to the JPDR-1 study since core average burnups were 3176 and 4400 MWD/MTUH and core average initial U-235 enrichments were 2.63 and 2.60 w/o for TMI-2 and JPDR-1 respectively.

A comparison of predicted vs chemical values for U-235 and plutonium was made on discharged Yankee-Rowe fuel.[8] Samples of dissolved fuel batches at the Nuclear Fuel Services plant in West Valley, New York were assayed chemically. Results are summarized by Table D5.

In practice, chemically standardized code values would be applied as item accountability directly to intact fuel assemblies where damage is not evident (or very minimal). The standarized code can be used as supporting measurements to weighing or NDA for fractional fuel assemblies from which fuel may have been dispersed throughout the pressure vessel and cooling system. Once the fractional fuel weight of a damaged fuel assembly has been established, for example, quantitative code values for the particular assembly can be applied as fractions of known initial values for fuel assembly weights. (This factor is referred to as "Pfr" in definitions given by Table D2.)

Core samples taken to substantiate PWR code predictability should be fuel material free of cladding, etc. Assays should include burnup, total uranium and plutonium, isotopic uranium and plutonium. The recommended technique is a triple spike mass spectrometric isotope dilution analysis based on a mixture of the isotopes U-233, Pu-242, and Nd-150 as spikes.

Isotope dilution analysis was first utilized on a production basis for uranium at the Idaho Chemical Processing Plant in the recovery process for spent U-235 reactor fuel.[13]

The intensity of an ion beam for each isotope as measured with a mass spectrometer is proportional to the isotope in the sample. The relationship used to determine total element by the isotope dilution technique is:

$$Cx = \frac{CsVs}{Vx}\left(\frac{A}{P} - 1\right) \qquad (D17)$$

where:

Cx = concentration of total element in the unknown solution
Cs = concentration of element in the added isotope (spike) solution
Vx = volume of the unknown solution
Vs = volume of spike solution
A = fractional isotopic purity factor for the spike isotope
P = fraction of the spike isotope in the mixture of sample and spike

The constant A is evaluated from an isotopic analysis of the particular spike material in use. P is determined from an isotopic analysis for each spiked sample. Since isotope dilution analysis is a function of relative intensities for each isotope in the sample. The technique enables evaluation of isotopic fractions as well as quantity of each isotope in the sample. By means of the triple spike technique, all of the uranium and plutonium values of interest as well as burnup analyses are available. Nd-150 serves as the spike to determine quantity of the burnup monitor Nd-148, or the Nd-145/Nd-146 ratio which also serves as a burnup monitor can be used. Atom % burnup based on Nd-148 is determined by:

$$B_{Nd} = \frac{100 \quad Nd/Y_{Nd}}{N_U + N_{Pu} + N_{Nd}/Y_{Nd}} \tag{D18}$$

where:

B_{Nd} is burnup in atom %

N_U N_{Pu} and N_{Nd} are numbers of atoms of uranium, plutonium, and neodymdium

Y_{Nd} is the cumulative fission yield of Nd-148 (0.0169)

In addition to total burnup analysis (U-235, Pu) as a function of the burnup monitor, the technique enables evaluation of U-235 depletion directly since total U-235 per unit fuel weight is known for initial fuel. Quantitative U-235 analysis in addition to the other isotopes of uranium and plutonium is of extreme importance in isotope dilution analysis.

Chemical separation techniques are employed to isolate uranium, plutonium and neodymium for assays by surface ionization mass spectrometry.[14,15] Nd-148 is a near ideal burnup monitor for thermal light water reactor fuels because (1) neodymium possesses the desirable chemical characteristics and behavior, both in the fuel and in solution, (2) its fission yield is essentially identical for U-235 and Pu-239 thermal fission, and (3) it has excellent properties for assay by surface ionization mass spectrometry. One

disadvantage Nd-148 may have as a burnup monitor is that results may be biased high due to the thermal neutron capture cross section of Nd-147 producing Nd-148.[16] The capture cross section for Nd-147 has been reported as 440 $\pm$ 150b.[17] It is not believed that Nd-148 production from Nd-147 neutron capture would significantly affect the low burnup values for TMI-2; however, the Nd-145/Nd-146 ratio can be evaluated for comparison.

The dominant error in burnup analysis lies with the value for the effective fission yield. In order to attain an uncertainty of 1.5 to 2.0% in burnup, requires that the uncertainty in the effective fission yield be in the range 1 to 1.5%. A review of fission product nuclear data for investigation of irradiated nuclear fuel burnup is described.[16]

Ratios of fissile isotopes have also been investigated as burnup monitors.[18] Various ratios of plutonium and uranium isotopes and combinations of isotopes were studied. Some are sensitive to enrichment or core location. Others are functions of burnup only. Another isotope of special interest is Ru-106 which can serve as a plutonium monitor since its yield from plutonium fission is 11 times that from uranium fission.

DISPERSED FUEL AND WASTE ASSAY

As a result of the reactor coolant pump being operated after damage occurred to the core, it is possible that fuel was distributed outside of the reactor pressure vessel by the coolant flow. The areas where fuel debris may have been deposited are taken from Reference 19 and are shown in Figures A4, A5 and D6.

- Once-through steam generator (upper tubesheet, lower dome area
- Reactor pressure vessel floor and a plenum
- Dead legs such as drains and letdown lines
- Reactor coolant pump
- Pressurizer (spray valve, vent valve, electromagnetic relief valve)
- Reactor coolant drain tank
- Makeup system (letdown line, letdown coolers, letdown orifice, seal return line)

During core removal it is likely that additional fuel material may fall to the bottom of the pressure vessel so that monitoring this area by instrumentation such as acoustic sensors may be desirable.

Dispersed Fuel Material

The dispersed fuel will exist in small quantities at various locations in the coolant systems away from its source. Therefore, the original enrichment and the present chemical composition (mainly with respect to the fuel-to-metal ratio) would be unknown unless the material can be sampled. Because of the small quantity of dispersed material expected (up to 5,000 kg U), the storage of representative samples in a hot cell off site for future analysis, may be feasible.

At the reactor site the visual examination (by remote TV) to make a preliminary estimate of the chemical composition of the fuel material and a weight measurement would permit a preliminary assessment of its SNM content after a neutron count of the material has been made.

For small quantities of material, small capacity load cells or scales and balances of the appropriate capacity would be necessary to obtain the greatest accuracy.

Changes in the core chemical composition can change the weight: the oxidation of the Zircaloy to ZrO_2 (6% error), oxidation of UO_{2+x} (error in all the UO_2 oxidizing to UO_3 is 6%), reaction of UO_2 hydrating to $UO_3 \cdot 2H_2O$, schoepite is 27%) and the retention of water in the interstices of the pile of rubble and as absorbed water adhering to the fuel material surface (no estimate of weight increase available). By not compensating for these effects, systematic errors in the mass measurement will occur. By analyzing the Zircaloy cladding and UO_2 for oxygen, this systematic error can be minimized.

Waste

Waste and unrecoverable fuel material may be the most difficult to measure, but because they are expected to comprise .03 to 0.6 percent of the total uranium, the measurement error may not make a significant contribution to the error in the material balance. The largest quantity of waste may come from fuel material which dissolved in the water and precipitated on the internal surfaces in the core as hydrated uranium oxide with the mineral name schoepite ($UO_3 \cdot XH_2O$).

The waste material can exist in several forms--uranium dissolved and suspended in water, uranium in the reactor coolant filters, and letdown systems and plate-out of uranium containing solids on the surfaces in the coolant system and particularly in the reactor pressure vessel.

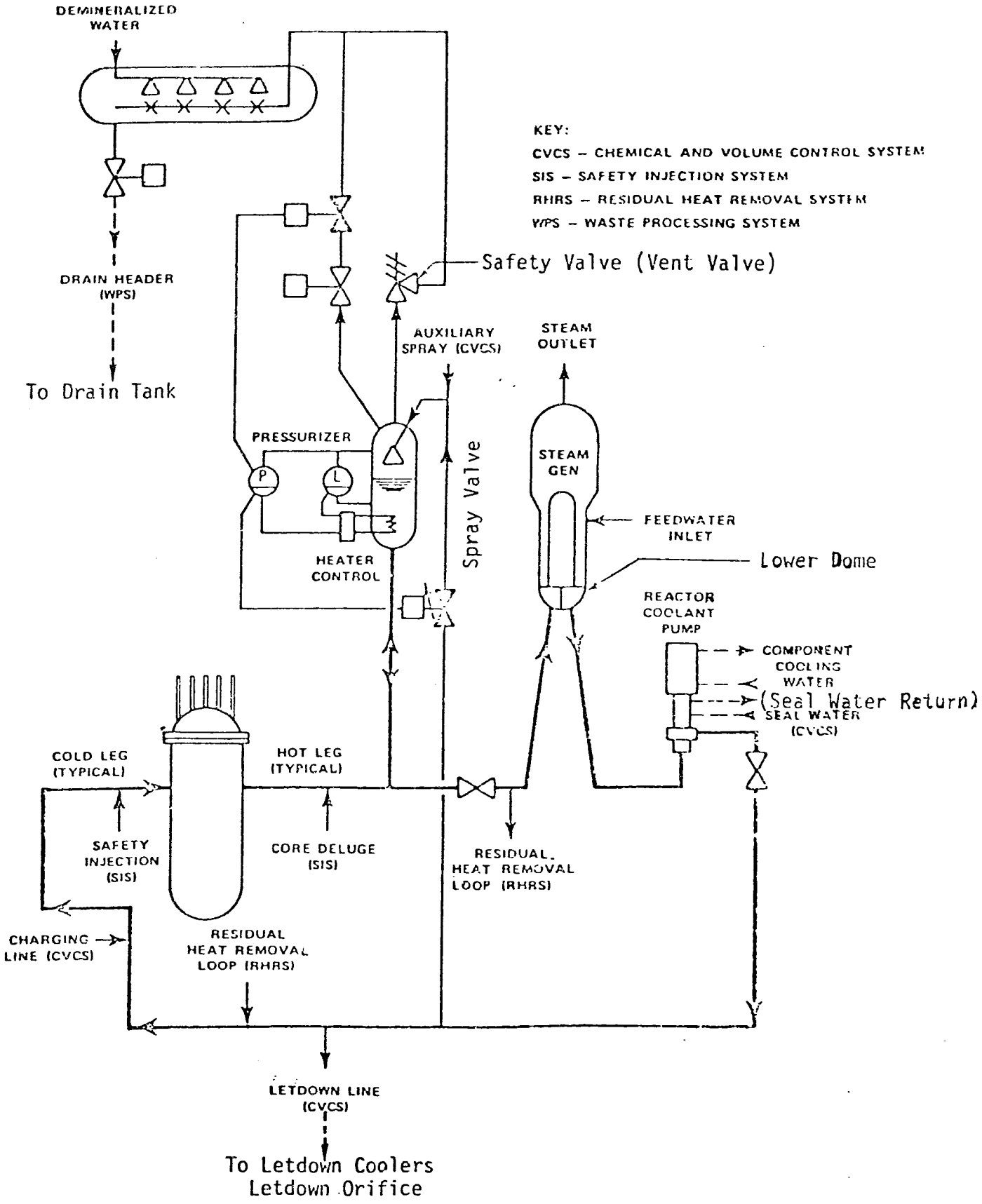


Figure D6. Reactor Coolant System, Flow Diagram

The volume of coolant containing uranium and plutonium can be measured by transferring the water to calibrated tanks in which the liquid level is measured by one of the methods such as the differential pressure technique discussed below under volume measurements. The elemental and isotopic composition of the coolant can be measured by using the isotopic dilution technique on a sample of the coolant.

The quantity of fuel material in the coolant filters can be obtained by weighing the filters and comparing the weight with the initial weight. The composition of the retained material could be determined by neutron counting techniques discussed under NDA, or by sampling the material collected on the filter and doing an isotopic dilution analysis as discussed previously.

Fine Particulate

A small quantity of the highly damaged fuel may exist in the form of fine particles. Since the reactor coolant pumps were operated for a period of time after damage to the core occurred, some of the particulate material could have been carried by the flowing coolant during the period that the reactor coolant pump was operating.

A worst case estimate of the volumetric fraction of fine fragments has been made by D. A. Powers (Appendix II.7 of Reference 3) in which it was found that 6% of the total core inventory may exist as fines less than .06 cm in diameter. Therefore, the quantity of dispersed material could be .06 x 82.9 x 10^3 kg (initial core inventory of uranium) = 5000 kg U dispersed in the coolant system.

Fuel-Water Reaction

Frequent sampling and chemical analysis at Sandia Laboratories of the coolant from the pressure vessel has not shown the presence of uranium or plutonium (at least at the parts per billion level which is the limit of the analytical measurement); however, fission products have been measured.

Likewise, no uranium and plutonium could be found in coolant samples taken from the steam generator and pressurizer. The pH of the coolant which is in the range of 8.1 - 8.4[2] (because of the addition of sodium hydroxide) may partially account for these results.

Since these results are not fully consistent with what would be expected from the leaching studies on uranium dioxide performed by Katayama[35] and on the solubility of uranium in water[21] a more detailed examination of the potential quantities of uranium and plutonium which may have reacted with the water is given below.

Under the assumed conditions of fuel lying in hot (but below 212°F) water in the reactor pressure vessel, over several years, the water can react with the fuel causing it to go into solution and precipitate as a hydrate on internal surfaces of the core and reactor pressure vessel.

In borated water at 195°F having a pH = 5.8[22] in which radiolysis has created oxidants, the concentration of uranium in a saturated solution can be 1 ppm[21]. If the solid suspension of hydrated uranium oxide ($UO_3 XH_2O$ - schoepite) is assumed to exist in the solution, the concentration of uranium is raised by a factor of 10[23] to 10 ppm U. There are 26,000 gallons of water in the TMI-2 reactor pressure vessel[19], so that the weight of uranium dissolved and suspended in the water can be calculated and is approximately 1000 grams U. For uranium enriched with 2.32 w/o U-235[5] this corresponds to 232 grams U-235.

The quantity of hydrated fuel material that could have dissolved and plated out can be calculated using the data from Appendix 2 in Reference 3 and an expression fitted to the leach data of Katayama[35] for the total amount leached. Using Katayama's leach rate expression, it is estimated that over a three-year-period (which is the time the core is scheduled to remain in place before removal operations begin) that 477 kg fuel could have been leached and plated out on the internal surfaces of the pressure vessel, vessel components, and fuel rods which are under water. This number seems

very large (probably because the leach rate is for leachant which is frequently and continuously changed and fresh leachant added) so that a more realistic number might be three to four times the quantity of uranium in the saturated solution or three to four kg uranium.

Fuel material in the letdown systems which is loose could be collected from the pipes, pumps, and valves where it may reside, be weighed, and analyzed by analytical chemical techniques or NDA techniques. The quantity of plated-out material on the inside of the pressure vessel could be determined by measuring the thickness and density of the deposit and analyzing for the concentration of uranium and plutonium. To obtain the total quantity of material plated out, an estimate of the surface area coated would have to be made. Nondestructive techniques for measuring the plating thickness such as eddy current and ultrasonic methods may be feasible. The selection of a particular technique would depend on the level of radioactivity in the RPV and the thickness and physical integrity of the coating.

Volume Measurements

It may be necessary to measure the volume and SNM concentration of large quantities of liquid to determine the weight of fissile material in some TMI-2 waste.

There are several ways that the liquid volume may be measured and these are discussed below:

Almost all bulk measuring techniques for determining the volume of liquid in a process tank rely on measuring the liquid level to indirectly obtain the volume of liquid. The simplest way to accurately measure the liquid level is with a sight glass attached to the side of a calibrated tank. This technique is practical when the radioactivity of the liquid is low enough so that personnel can work near the tank. However, when isolation of the process tank is necessary because of the hazardous level of radioactivity of the tank contents, the liquid level sensor may be separated from the readout

instrument by several walls of concrete. A commonly used method in this situation is the differential pressure gauge or liquid manometer. This type of device may be interfaced with a computer, is accurate, and can be installed in a central control room. These advantages may make the use of the differential pressure gauges most favorable, even when other techniques are feasible.

Differential Pressure Gauge

The differential pressure gauge measures the difference in hydrostatic pressure between two positions, one of which is in the liquid being measured and the other in the vapor space directly above the liquid. The difference in pressure divided by the density of the liquid (which can be measured when the openings of both dip tubes are in the water a fixed distance apart) is the distance between the opening of the lower dip tube and the top of the liquid in the tank. The pressure sensor can be a manometer or a pressure gauge. The overall error in measuring the liquid level by the differential pressure technique is $\sim$ 0.2%.

Time Domain Reflectometry

Time domain reflectometry (TDR) is another technique for measuring the liquid level.[25]

TDR is often referred to as a "closed-loop-radar" system. The system operates by sending a voltage pulse down a transmission probe which is immersed in the liquid. When the pulse encounters any deviation in impedance (which includes the dielectric constant of the material), such as at the vapor-liquid interface, part of the incident pulse is reflected back to the sending point, where its trip time and its amplitude are compared with the original pulse. The amplitude of the received pulse is measured as the ratio of the amplitude of the reflected signal to the amplitude of the transmitted signal. This ratio, called the reflection coefficient, is a function of the

di-electric constant of the coaxial line insulator. The reflection co-efficient of the TDR probe may be displayed on a CRT screen as a parameter in the "y" direction. A rapid change in slope along the "x" axis corresponds to the point of the liquid-vapor interface on the TDR probe. The location of the change in slope is used to obtain the liquid level in the tank. The average TDR instrument liquid level resolution is on the order of 1 to 2 mm.

The TDR system is attractive for safeguards applications because any tampering with the system (e.g., placing a hole in the probe's outer conductor to give a false reading) can be easily identified. Advantages of the TDR system are that it can be interfaced to a computer and that the TDR probe can be designed to give temperature, pressure, and liquid level measurements simultaneously.

Isotopic Dilution/Volume Measurements

The volume of liquid in a process tank can be determined by adding a known amount of a stable tracer isotope[25,26] and measuring a sample of the solution by mass spectrometry to determine the concentration of the addition.

Mass spectrometry is used in this method to accurately determine the ratio of the quantities of isotopes present in the sample, rather than measuring their absolute quantities.

To measure the isotopic dilution in the process solution, a sample is withdrawn after the addition is made of a tracer having a certain isotopic ratio. The sample is then spiked with another isotope having a different isotopic ratio from that of the tracer. The ratios of the isotopes in the two solutions, before and after addition of the spikant to the solution, are used to determine the dilution factor and to calculate the volume of process solution.

A difference in volumes between the isotopic dilution technique and the differential pressure method of 0.34% has been reported.[25]

Miscellaneous Bulk Measurement Techniques

Other techniques for measuring the level of liquid in a tank are: conductivity probe, inductivity probe, resistance-wire elements, ultrasonic gauge and capacitive continuous-level transducers.[6,24]

Other Possible Analytical Methods

Most other analytical techniques require the separation of zirconium from the solution because of its interference. The separation can be carried out by dissolving the unknown solution of fuel material in hydrochloric acid and reducing the uranium to U^{+4} and plutonium to Pu^{+3} by using sulfur dioxide or hydroglamin. PuF_3 and UF_4 can be quantitatively precipitated from this solution leaving behind the ZrF_4 in solution by the addition of hydrofluoric acid. Another separation method uses anionite ion exchange resin and 8 M Hcl to quantitatively separate the zirconium from the uranium and plutonium fractions which are absorbed on the resin. Dissolution of uranium and plutonium in the presence of zirconium is described.[27]

Besides isotopic dilution methods, potential controlled coulometry may be used to analyze for elemental uranium,[28, 29, 30] and plutonium.[28, 30] In this method it is necessary to separate either the uranium (by Hexone extraction) or plutonium (by an anion exchange method[28, 29] so the coulometry can be performed with separate solutions. The error in coulometry is 0.3% to 0.6%.

Other methods[31] for analyzing the plutonium and uranium are the plutonium extraction from a hot solution of TTA and radiometric measurement of plutonium by alpha counting (using pulse height analysis).[30] These methods are not suitable because they give accuracies of about 3%. For this reason isotopic dilution is the analytical method that appears to be best suited for plutonium and uranium since it enables both elemental and isotopic assays in addition to burnup.

Considerations in the Analysis of TMI-2 Fuel

(a) Effects of Irradiation on NDA Analysis. There is no adverse effect of radiation on NDA analysis of separated samples because of γ-radiation or neutron flux in the irradiated material is used for detection and identification. If the radiation level in it is too high, it can be diminished by taking only a small sample for measurements.

(b) Effect of Radiation on Chemical Analysis. The methods used for chemical analysis-- the isotopic dilution of uranium and plutonium involves liquid--liquid extraction procedures for purification of the sample thereby eliminating the effects of radioactive impurities from direct mass interferences and reducing the biological hazard. The size of the sample can be made very small (since mass spectrometry needs only a few micrograms of sample on its ionization filament) so that any interfering emissions from the fission products can be reduced.

TRACKING SYSTEM FOR TMI-2 RECOVERED FUEL

Intact and breached fuel components, dispersed fuel material, samples and waste materials will all have to be accounted for and tracked until final analyses, recovery, or disposition has been completed. Since these various elements will be dispersed over a large geographical area and time span, tracking presents a complex and difficult situation. A method or philosophy must be established prior to start of the fuel recovery operation.

Several methods are currently being employed in today's accountability systems for fuel materials and components. But unfortunately most of these systems deal mainly with well characterized materials and unique fuel components. This, for the most part, will not be the case for the TMI-2 core.

Tracking System

Tracking of the various recovered core elements can best be accomplished by use of a follower card methodology. The following discussion outlines this method: Follower cards in the form of key punch data cards are to be generated, in duplicate, for each canister, cask, drum or container used for containing core component materials. These cards will then be used on a computer system to generate the necessary information and reports required for tracking and accountability. One card is to be maintained at the central accountability center for TMI-2 fuel and the other card to reside with the container and its contents. Each time a container is moved, the residing container follower card will accompany it. If the contents of the container are modified, i.e., sampled, split, transferred, etc, then new follower cards will have to be generated for both the accountability center and the container reflecting the modification, and entered into the TMI-2 tracking system.

Follower Card Information

The follower cards should contain information on: Container number, type of container, core component number (if available), sample number (if applicable), type of core component (if available), core sector location (estimated), material form (whole assembly, sector cuttings, loose pellets, waste, etc), material type (uranium, plutonium if present), gross weight, net weight, element weight (estimated), weight % isotope (estimated), and isotope weight (estimated).

Obviously all this information cannot fit on a single 80 sector card so two or more cards will have to be used per container.

Container Identification

Unique container identification numbers will have to be assigned and permanently fixed to each container. This number may all too often be the only physical identification for inventorying and tracking core materials. This number also serves as a key sorting item on the keypunch cards for producing inventory and material control records.

Core Material Transfers

Core materials that are transferred to various facilities for analyses, study, recovery, etc, should be tracked as well. Each transfer requires documentation to be prepared. The follower cards will greatly assist in this effort by producing information for the documentation and aid as backup data for the recipient.

The existing container follower card should accompany the shipment and be maintained with the core material until final disposition is made. If the core material becomes separated from its original container, due to the need for reuse of that container, new follower cards should be issued deleting the container number. A new numbering system should then be employed to help identify the material.

Final Disposition of Follower Cards

When final disposition of core material has been completed (i.e., sent to burial or permanent storage), the container follower card should be returned to the TMI-2 accountability center. This will indicate the material can now be removed from the tracking system.

Advantages and Disadvantages

As with any system there are advantages and disadvantages. Some of these will be discussed.

Advantages

- The system is a simple method for tracking material wherever it may reside.
- All the information for accountability requirements is listed in one location and easy to access.
- The residing container cards aid in backup information for the recipient of the material.
- Duplication of the keypunch cards is easily accomplished.
- Storage of the keypunch cards requires minimal space.
- Use of the keypunch cards reduces transcription errors.

Disadvantages

- Organizations receiving the keypunch cards have a computer system capable of inputting the cards or inputting the data manually.
- The cards are fragile and easily damaged.
- The cards can easily become separated or lost from the shipment.
- Organizations having to make changes to the cards must have keypunching capabilities or employ a service.

SECTION E

REFERENCES

1. Final Report of SL-1 Recovery Operation, SL-1 Project Idaho Test Station, General Electric Company, July 17, 1962. IDO 19311.

2. Personal communication with D. A. Powers at Sandia Laboratory, Albuquerque, New Mexico, on September 23, 1980.

3. D. A. Powers, Sandia Laboratories, "Status of the Reactor Core Based on Fission Product Analysis", November 29, 1979.

4 Output for TMI-2 Fuel from PDQ Code Received from Babcock and Wilcox Company, Lynchburgh, Virginia, September 23, 1980.

5. R. G. Canada, "NSAC EPRI ORIGEN Code Calculation of TMI-2 Fission Product Inventory", Report No. R-80-012 by Technology for Energy Corporation, Knoxville, Tennessee, May 2, 1980.

6. A. Sugarman, (co-author) et al, "Handbook of Safeguards Measurement Methods", by Mound Laboratory. To be published by the Nuclear Regulatory Commission, Office of Standards Development.

7. Personal communication with C. Vaughan, General Electric Company, Wilmington, North Carolina, on January 19, 1979.

8. R. J. Cacciapouti, "Evaluation of the Yankee Method for Calculating Uranium Depletion and Plutonium Production in Exposed Fuel", YAEC 1060, 1972.

9. T. Gozani, "Active Direct Measurement of Residual Fissile Content in Spent Fuel Assemblies", EPRI Research Report 278-1, July 1975.

10. S. T. Hsue, et al, "Nondestructive Assay Methods for Irradiated Nuclear Fuels", LA-6923, January 1978.

11. S. T. Hsue, et al, "Passive Nuclear Assay of Irradiated Nuclear Fuels", LA-7645-MS, February 1979.

12. H. Natzume, et al, "Gamma-Ray Spectrometry and Chemical Analysis Data of JPDR-1 Spent Fuel", Journal of Nuclear Science and Technology, 14 10 , pp 745-761, October 1977.

13. Paul Goris, et al, "Uranium Determination by the Isotope Dilution Technique", Anal. Chem., Vol. 29, p 1590, November 1957.

14. S. F. Marsh, M. R. Orty, R. M. Abernathy, J. E. Rein, LA-5568, "Improved Two-Column Ion Exchange Separation of Plutonium, Uranium, and Neodymium in Mixed Urnaium-Plutonium Fuels for Burnup Measurements", June 1974.

15. W. J. Maeck, R. P. Larsen, J. E. Rein, TID 26209, Burnup Determination for Fast Reactor Fuels", February 1975

16. W. J. Maeck, ICPP-1160, "A Review of Fission Product Nuclear Data Requirements for Investigation of Irradiated Nuclear Fuel", p 3, July 1978.

17. D. Heck, et al, "Atomkernenergie 24, 141, 1974.

18. G. P. Selby, B. A. Napier, C. L. Timmerman, ISPD-37, PNL-2761, "Selected Isotopic Functions: A Description and Demonstration of Their Uses", October 1978.

19. Planning Study for Phase II - Reactor Internals, Disassembly, Fuel Removal, and Systems Cleanup, Bechtel Power Corporation, May 28, 1980.

20. Personal discussions with W. R. Gibson, Babcock and Wilcox, Lynchburg, Virginia, September 9, 1980.

21. H. D. Holland and L. H. Brush, "Uranium Oxides in Ores and Spent Fuels", delivered at High-Level Radioactive Solid Waste Forms, An In-Depth Technical Conference, sponsored by the U. S. Nuclear Regulatory Commission, Denver, Colorado, December 19-21, 1978.

22. Y. Solomon and J. Roesner, "Some Observations on the Possible Relationship of Reactor Coolant Chemistry and Radiation Level Buildup", WCAP-9407, November, 1979.

23. Personal Discussions with Ron Wang at Hanford Engineering Development Laboratory on September 16, 1980.

24. D. Nentwick, et al, "R&D Work on Safeguarding the Input to a Fuel Reprocessing Plant (Accountability Task), Institute for Angewandte Systemtechnik and Reaktorphysik, Kernforgoschungszentrum, Karlsruhe, Germany, KFK 1700.

25. C. K. Mathews, et al, "An Independent Method for Input Accountability in Reprocessing Plants: Magnesium Tracer Technique for the Accountability of Plutonium (MAFTRAP)", BARC-809, 1975.

26. H. Bokeland, "Investigation of Reprocessing Input Measurement Using Tracer Technique:, Eurochemic Technical Report, No. 266, NP 18577. October 1970.

27. Rose Marie Gonzales de Duval, Joel W. Dahlby, Adrian P lovell, LA-8533 "Application of the Sealed-Reflux Dissolution System to 100-Gram Samples.

28. 1974 Annual Book of ASTM Standards, Part 45, Nuclear Standards American Society of Testing and Materials, Philadelphia, 1974.

29. C. J. Rodden, "Selected Methods for Plutonium and Uranium in the Nuclear Fuel Cycle", USAEC 1972, Second Edition.

30. B. B. Stephen, et al, "Methods for the Accountability of Mixed Oxide", Lawrence Livermore Laboratory, June 1977.

31. T. Gozani, J. Maly, "A Study of Nuclear Material Accountability:, NUREG-2090, Volume 3, LBL and NRC, June 1977.

32. M. Rogovin, "A Report to the Commissioners and to the Public", Volume II, Part 2, January 1980.

33. M. L. Picklesimer, "Bounding Estimates of Damage to Zircaloy Fuel Rod Cladding in the TMI-2 Core at Three Hours After the Start of the Accident:, March 28, 1979. Memo to File of U. S. Nuclear Regulatory Commission, June 20, 1979.

34. Robert T. Anderson, "Scoping Studies of the Alternate Options for Defueling, Packaging, Shipping, and Disposing of the TMI-2 Spent Fuel Core," AGNS-35900-1.5-79, September 1980.

35. Y. B. Katayama and J. E. Mendel, "Leaching of Irradiating LWR Fuel Pellets in Dionized Water, Sea Brine, and Typical Ground Water", ANS Trans., 7:447, November-December, 1977.

36. T. J. Thompson and J. G. Beckerley, "The Technology of Nuclear Reactor Safety", Volume I, the MIT Press, Massachusetts Institute of Technology, Cambridge, Massachusetts, 1964.

37. W. J. Freede, "SRE Core Recovery Program", NAA-SR-6359.

38. D. G. Hurst, "The Accident to the NRX Reactor, Part II", AECL-233, October 1953.

39. A. F. Rupp, "NRU Reactor Incident, Nuclear Safety, 1, 3, Page 70, 1960"

40. R. B. Korsmeyer, "Westinghouse Testing Reactor Incident", Nuclear Safety, 2, 2 p. 70 (1960).

41. L. A. Mann, "EBR-1 Operating Experience", Nuclear Safety, 3, 2, p. 68 (1961).

42. John Grahm, "Fast Reactor Safety", Academic Press, New York (1971).

43. E. L. Alexanderson, et al, "October 5, 1966 - Fuel Damage Incident at the Enrico Fermi Atomic Power Plant, Status as of February 24, 1967", NP 16750.

APPENDIX 1

10CFR70.51
MATERIAL BALANCE, INVENTORY, AND RECORD REQUIREMENTS

DOE 5630.2
CONTROL AND ACCOUNTABILITY OF NUCLEAR MATERIALS
BASIC PRINCIPLES

APPENDIX 1

10CFR70.51
MATERIAL BALANCE, INVENTORY, AND RECORDS REQUIREMENTS

[38 FR 33970, Dec. 10, 1973, as amended at 40 FR 8792, Mar. 3, 1975; 43 FR 6925, Feb. 21, 1978]

§ 70.44 Creditor regulations.

(a) Pursuant to section 184 of the Act, the Commission consents, without individual application, to the creation of any mortgage, pledge, or other lien upon any special nuclear material, not owned by the United States, which is subject to licensing: *Provided:*

(1) That the rights of any creditor so secured may be exercised only in compliance with and subject to the same requirements and restrictions as would apply to the licensee pursuant to the provisions of the license, the Atomic Energy Act of 1954, as amended, and regulations issued by the Commission pursuant to said act; and

(2) That no creditor so secured may take possession of the special nuclear material pursuant to the provisions of this section prior to either the issuance of a license by the Commission authorizing such possession or the transfer of a license pursuant to § 70.36.

(b) Nothing contained in this section shall be deemed to affect the means of acquiring, or the priority of, any tax lien or other lien provided by law.

(c) As used in this section, "creditor" includes, without implied limitation, the trustee under any mortgage, pledge, or lien on special nuclear material made to secure any creditor, any trustee or receiver of the special nuclear material appointed by a court of competent jurisdiction in any action brought for the benefit of any creditor secured by such mortgage, pledge, or lien, any purchaser of such special nuclear material at the sale thereof upon foreclosure of such mortgage, pledge, or lien or upon exercise of any power of sale contained therein, or any assignee of any such purchaser.

(Sec. 184, 68 Stat. 954, as amended; 42 U.S.C. 2234)

[32 FR 2563, Feb. 7, 1967, as amended at 35 FR 11461, July 17, 1970]

SPECIAL NUCLEAR MATERIAL CONTROL, RECORDS, REPORTS AND INSPECTIONS

§ 70.51 Material balance, inventory, and records requirements.

(a) As used in this section:

(1) "Additions to material in process" means receipts that are opened except for receipts opened only for sampling and subsequently maintained under tamper-safing, and opened sealed sources.

(2) "Enrichment category" for uranium-235 means high-enriched uranium—that uranium whose isotope content is 20 percent or more uranium-235 by weight, and low-enriched uranium—that uranium whose isotope content is less than 20 percent uranium-235 by weight.

(3) "Element" means uranium or plutonium.

(4) "Fissile isotope" means (i) uranium-233 or (ii) uranium-235 by enrichment category.

(5) "Limit of error" means the uncertainty component used in constructing a 95 percent confidence interval associated with a quantity after any recognized bias has been eliminated or its effect accounted for.

(6) "Material balance" means a determination of material unaccounted for (MUF) by subtracting ending inventory (EI) plus removals (R) from beginning inventory (BI) plus additions to inventory (A). Mathematically,

$$MUF = BI + A - EI - R$$

(7) "Material in process" means any special nuclear material possessed by the licensee except in unopened receipts, sealed sources, and ultimate product maintained under tamper-safing.

(8) "Physical inventory" means determination on a measured basis of the quantity of special nuclear material on hand at a given time. The methods of physical inventory and associated measurements will vary depending on the material to be inventoried and the process involved.[1]

(9) "Removals from material in process" includes measured quantities of

[1] Criteria for physical inventories are set out in paragraph (f) of this section.

special nuclear material disposed of as discards, encapsulated as a sealed source, or in other ultimate product placed under tamper-safing or shipped offsite.

(10) "Tamper-safing" means the use of devices on containers or vaults in a manner and at a time that ensures a clear indication of any violation of the integrity of previously made measurements of special nuclear material within the container or vault.

(11) "Ultimate product" means any special nuclear material in the form of a product that would not be further processed at that licensed location.

(12) "Unopened receipts" means receipts not opened by the licensee, including receipts of sealed sources, and receipts opened only for sampling and subsequently maintained under tamper-safing.

(b)(1) Each licensee shall keep records showing the receipt, inventory (including location), disposal, acquisition, and transfer of all special nuclear material in his possession regardless of its origin or method of acquisition.

(2) Records which are required by the regulations in this part or by license condition shall be maintained for the period specified by the appropriate regulation or license condition. If a retention period is not otherwise specified by regulation or license condition, such records shall be maintained until the Commission authorizes their disposition.

(3) Records of receipt, acquisition, or physical inventory, of special nuclear material which must be maintained pursuant to paragraph (b)(1) of this section shall be maintained as long as the licensee retains possession of the material and for five years following transfer of such material. Records of inventory maintained to demonstrate compliance with paragraph 70.58(h) shall be maintained for six months.

(4) [Reserved]

(5) Records of transfer of special nuclear material to other persons shall be maintained by the licensee who transferred the material until the Commission authorizes their disposition. Records required by paragraph (e)(1)(v) of this section shall be maintained for five years.

(6) Records of disposal of special nuclear material shall be maintained in accordance with § 20.401(c) of this chapter.

(c) Each licensee who is authorized to possess at any one time special nuclear material in a quantity exceeding one effective kilogram of special nuclear material shall establish, maintain, and follow written material control and accounting procedures which are sufficient to enable the licensee to account for the special nuclear material in his possession under license.

(d) Except as required by paragraph (e) of this section, each licensee who is authorized to possess at any one time and location special nuclear material in a quantity totaling more than 350 grams of contained uranium-235, uranium-233, or plutonium, or any combination thereof, shall conduct a physical inventory of all special nuclear material in his possession under license at intervals not to exceed twelve months.

(e) Effective May 6, 1974, each licensee who is authorized to possess at any one time special nuclear material in a quantity exceeding one effective kilogram of special nuclear material and to use such special nuclear material for activities other than those involved in the operation of a nuclear reactor licensed pursuant to Part 50 of this chapter or those involved in a waste disposal operation; as sealed sources; or as reactor irradiated fuels involved in research, development, and evaluation programs in facilities other than irradiated fuel reprocessing plants, shall:

(1) Maintain procedures which shall include:

(i) Procedures for tamper-safing containers or vaults containing special nuclear material not in process, which include control of access to the devices and records of the date and time of application of each device to a container or vault; unique identification of each such item; inventory records showing the identity, location, and quantity of special nuclear material for all such items; and records of the source and disposition of all such items;

(ii) Records of the quantities of special nuclear material added to or removed from the process;

(iii) Inventory records for the quantity of special nuclear material in process;

(iv) Unique identification of items or containers containing special nuclear material in process; inventory records showing the identity, location, and quantity of special nuclear material for all such items; and records of the source and disposition of all such items;

(v) Documentation of all transfers of special nuclear material between material balance areas to show identity and quantity of special nuclear material transferred;

(vi) Requirements for authorized signatures on each document for transfer of special nuclear material between material balance areas; and

(vii) Means for control of and accounting for internal transfer documents.

(2) On or before May 6, 1974, and thereafter as necessary to comply with the requirements of paragraph (e)(3) of this section, perform a physical inventory of all special nuclear material in his possession in compliance with the criteria for physical inventories set forth in paragraph (f) of this section.

(3) Conduct physical inventories made in accordance with the criteria for physical inventories set forth in paragraph (f) of this section at intervals determined from the start of the beginning inventory to the start of the ending inventory not to exceed:

(i) 2 calendar months for plutonium except for plutonium containing 80 percent or more by weight of the isotope Pu-238, uranium-233 and for uranium enriched 20 percent or more in the isotope uranium-235 (except as provided in paragraph (e)(3)(ii) of this section); and

(ii) 6 calendar months for uranium enriched less than 20 percent in the isotope uranium-235; for plutonium, U-233 and high-enriched uranium in that portion of an irradiated-fuel reprocessing plant from the dissolver to the first vessel outside of the radiation shielded portion of the process; and for plutonium containing 80 percent or more by weight of the isotope Pu-238;

(4) Within 30 calendar days after the start of each ending physical inventory required by paragraph (e)(3) of this section:

(i) Calculate, for the material balance interval terminated by that inventory, the material unaccounted for (MUF) and its associated limit of error for each element and the fissile isotope for uranium contained in material in process;

(ii) Reconcile and adjust the book record of quantity of element and fissile isotope, as appropriate, to the results of the physical inventory;

(iii) Complete and maintain for a period of five years material balance records for each material balance showing the quantity of element and fissile isotope, as appropriate, in each component of the material balance with the associated limit of error for the material unaccounted for both in terms of absolute quantity of element and fissile isotope and relative to additions to or removals from material in process for the interval, where results of limit of error calculations are recorded in sufficient detail to permit an evaluation of sources of error.

(iv) Complete and maintain for a period of five years a record summarizing the quantities of element and fissile isotope, as appropriate, for ending inventory of material in process, additions to material in process during the material balance interval and removals from the material in process during the material balance interval; and

(v) Complete and maintain for a period of five years a record summarizing the quantities of element and fissile isotope, as appropriate, in unopened receipts (including receipts opened only for sampling and subsequently maintained under tamper-safing), and ultimate products maintained under tamper-safing, or in the form of sealed sources;

(5) Establish and maintain a system of control and accountability such that the limits of error for any material unaccounted for (MUF) ascertained as a result of the material balances made pursuant to paragraph (e)(3) of this section do not exceed (i) 200 grams of plutonium or uranium-233, 300 grams of high enriched uranium

or uranium-235 contained in high enriched uranium, or 9,000 grams of uranium-235 contained in low enriched uranium, (ii) those limits specified in the following table, or (iii) other limits authorized by the Commission pursuant to paragraph (e)(6) of this section:

Material Type	*Limit of Error of MUF on Any Total Plant Inprocess Material Balance*[1] *Percent*
Plutonium element or uranium-233 in a chemical reprocessing plant	1.0
Uranium element and fissile isotope in a reprocessing plant	0.7
Plutonium element, uranium-233, or high enriched uranium element and fissile isotope—all other	0.5
Low-enriched uranium element and fissile isotope—all other	0.5

[1]As a percentage of additions to or removals from material in process, whichever is greater.

Any licensee subject to this paragraph on December 6, 1973, who requests higher limits pursuant to paragraph (e)(6) of this section at the time he submits his program description under the provisions of paragraph (g) of this section is hereby authorized to operate at the higher limits until the application for license or amendment has been finally determined by the Commission;

(6) An applicant or a licensee subject to the requirements of paragraph (e) of this section may request limits higher than those specified in paragraph (e)(5) of this section. The requested higher limits shall be based on considerations such as the type and complexity of process, the number of unit operations, process throughput quantities, process recycle quantities, and the technology available and applicable to the control and accounting of the material in the process. The Commission will approve higher limits if the applicant demonstrates:

(i) That he has made reasonable efforts and cannot meet the limits of error of MUF specified in paragraph (e)(5) of this section; and

(ii) That he has initiated or will initiate a program to achieve improvements in his material control system so as to meet the limits specified in paragraph (e)(5) of this section.

(f) Each licensee subject to the requirements of paragraph (e) of this section shall:

(1) Establish physical inventory procedures to assure that:

(i) The quantity of special nuclear material associated with each item on inventory is a measured value;

(ii) Each item on inventory is listed and identified to assure that all items are listed and that no item is listed more than once;

(iii) Cutoff procedures for transfers and processing are established so that all quantities are inventoried and none are inventoried more than once;

(iv) Cutoff procedures for records and reports are established so that all transfers for the inventory and material balance interval and no others are included in the records; and

(v) Upon completion of the inventory, all book and inventory records, both total plant and material balance area, are reconciled with and adjusted to the physical inventory.

(2) Establish inventory procedures for sealed sources and containers or vaults containing special nuclear material that provide for:

(i) Identification and location of all such items;

(ii) Verification of the integrity of the tamper-safing devices for such items;

(iii) Reverification of identity and quantity of contained special nuclear material for each item not tamper-safed, or whose tamper-safing is found to have been compromised;

(iv) Verification of the correctness of the inventory records of identity and location for all such items; and

(v) Documentation in compliance with the requirements of paragraphs (f)(2)(i), (ii), (iii), and (iv) of this section.

(3) Establish inventory procedures for special nuclear material in process that provide for:

(i) Measurement of all quantities not previously measured by the licensee for element and fissile isotope; and

(ii) For all material whose content of element and fissile isotope has been previously measured by the licensee

U.S. Department of Energy

Washington, D.C.

ORDER

DOE 5630.2

8-21-80

SUBJECT: CONTROL AND ACCOUNTABILITY OF NUCLEAR MATERIALS, BASIC PRINCIPLES

1. PURPOSE. This Order contains the basic principles and requirements for control and accountability of all nuclear materials. This Order, coupled with those in the Department of Energy's (DOE) 5630 series, together with physical security requirements outlined in the DOE 5632 series, comprise an overall integrated safeguards and security system for those nuclear materials meeting the definition for special nuclear material (SNM).

2. SCOPE. The provisions of this Order apply to those facilities involving nuclear operations under the management of the Assistant Secretary for Nuclear Energy, the Assistant Secretary for Resource Applications, the Director of Energy Research, and the Assistant Secretary for Defense Programs, and operations offices; and include all elements of the Department, their license-exempt contractors, and subcontractors which possess, use, or ship nuclear materials. Further, this Order applies to DOE-owned other nuclear material (e.g., deuterium, americium, curium) at licensed and unlicensed facilities when these materials are not covered by Nuclear Regulatory Commission (NRC) license.

3. BACKGROUND. This Order replaces Energy Research and Development Administration Manual Appendix 7401-A, SAFEGUARDS CONTROL AND MANAGEMENT OF NUCLEAR MATERIAL, PRINCIPLES AND GENERAL STANDARDS.

4. REFERENCES.

   a. DOE 5630.1, CONTROL AND ACCOUNTABILITY OF NUCLEAR MATERIALS, of 8-3-79, which establishes the responsibilities and authorities for material control and accountability.

   b. DOE 5632.2, PHYSICAL PROTECTION OF SPECIAL NUCLEAR MATERIALS, of 2-16-79, which establishes minimum physical protection standards for special nuclear materials.

   c. "Safeguards Analytical Laboratory Evaluation (SALE) 1977 Annual Report," Volume I, New Brunswick Laboratory report NBL-290, May 1978.

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INITIATED BY:
Office of Safeguards and Security

5. RESPONSIBILITIES AND AUTHORITIES. See DOE 5630.1, CONTROL AND ACCOUNTABILITY OF NUCLEAR MATERIALS, of 8-3-79.

William S. Heffelfinger
Director of Administration

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CHAPTER I

DEFINITIONS

1. ACCURACY of measurement indicates the agreement between the true value and th measured value. The "true" value is considered to be the best obtainable value and is arrived at usually by multiple measurement of standard or reference materials.

2. ALARM LIMITS are the established values for inventory differences which when exceeded require immediate action and reporting to the cognizant operations office and the Office of Safeguards and Security (DP-30). For processing, production, and fabrication operations, alarm limits will be established with a 99 percent probability.

3. APPARENT LOSS is the inability to locate physically or to otherwise account for:

   a. Any identifiable or discrete item (e.g., batch, lot, or piece) containing nuclear material.

   b. An inventory difference quantity of nuclear material where the book inventory is larger than the physical inventory by an amount which is in excess of the established alarm limit.

4. BOOK INVENTORY is the amount of material present at a given time as reflected by accounting records such as the general and subsidiary ledgers, i.e., the beginning physical inventory adjusted for receipts and removals for a given reporting period.

5. BULK MATERIALS are materials in any physical or chemical form which are not identifiable as discrete items and thus must be accounted for by weight, volume, sampling, and chemical analysis or nondestructive analysis.

6. CATEGORY I QUANTITIES OF SNM.

   a. Uranium 235 (contained in Uranium enriched to 20% or more in the isotope U-235) alone, or in combination with Plutonium and/or Uranium 233 when (multiplying the Plutonium and/or Uranium 233 content by 2.5) the total is 5,000 grams or more.

   b Plutonium and/or Uranium 233 when the Plutonium and/or Uranium 233 content is 2,000 grams or more.

7. CATEGORY II QUANTITIES OF SNM.

   a. Uranium 235 (contained in Uranium enriched to 20% or more in the isotope U-235) alone, or in combination with Plutonium and/or Uranium 233 when (multiplying the Plutonium and/or Uranium 233 content by 2.5) the total is 1,000 to 4,999 grams.

   b. Plutonium and/or Uranium 233 when the Plutonium and/or Uranium 233 content is 400 grams to 1,999 grams.

8. CATEGORY III QUANTITIES OF SNM.

   a. Uranium 235 (contained in Uranium enriched to 20% or more in the isotope U-235) when the total is 1 gram to 999 grams.

   b. Plutonium and/or Uranium 233 when the Plutonium and/or Uranium 233 content is 1 gram to 399 grams.

   c. Combinations of Plutonium and/or Uranium 233 with Uranium 235 (contained in Uranium enriched to 20% or more in the isotope U-235) when the total is less than 1,000 grams and the Plutonium and/or Uranium 233 content is less than 400 grams.

   d. Uranium 235 contained in Uranium enriched to less than 20% in the isotope U-235 in all quantities above .99 grams.

9. CATEGORY IV QUANTITIES OF SNM. Reportable quantities and above not covered by Categories I, II, or III.

10. CONFIRMATORY MEASUREMENT is a measurement made to test if some attribute or characteristic of the nuclear material is consistent with the expected response for that material if no change has occurred. A confirmatory measurement may include go/no-go, qualitative, semiquantitative, or verification measurements.

11. CONTROL LIMITS are the established values beyond which any variation, in this case inventory difference, is considered to be an indication of the presence of an assignable cause, and the variation should be investigated. Control limits should usually be established with a 95 percent probability and called warning limits, while those usually with a 99 percent probability are called alarm limits.

12. EQUIPMENT HOLDUP is an estimated or measured quantity of nuclear material which adheres so tenaciously to the equipment that it has become part of the equipment or requires special treatment to remove.

13. FACILITY is a generic term used to refer to an integral nuclear material handling complex, usually operated by a single contractor.

14. GRADED SAFEGUARDS is a system designed to provide varying degrees of physical protection, accountability, and material control to different types, quantities, and chemical composition, physical form and isotopic composition of SNM consistent with varying levels of attractiveness and convenience to possible adversaries.

15. IN-PROCESS INVENTORY refers to the quantity of nuclear material present in a fabrication or process line, in processing vessels and machines at any specified time.

16. INTERNAL CONTROL SYSTEM is a set of administrative and accounting policies and procedures implemented by a facility in order to account for and maintain control of nuclear material. It includes checks and balances in the division of duties so designed that the work of one will serve to verify the work of another.

17. INVENTORY DIFFERENCE (ID) is the algebraic difference between the nuclear material book inventory (BI) and a physical inventory (PI), i.e., ID = BI - PI.

18. IRRADIATED MATERIAL refers to nuclear material which has been exposed to radiation, as from a nuclear reactor, and as a consequence delivers an external radiation dose rate which requires special containment, handling, and measurement procedures because of the penetrating radiation levels.

19. LIMITS OF ERROR of an estimator T, as applied to this directive, is twice the standard deviation of T unless otherwise stipulated.

20. MATERIAL BALANCE AREA (MBA) is an identifiable physical area wherein the quantity of nuclear material being moved into or out is represented by a measured value.

21. MATERIAL CONTROL AND ACCOUNTABILITY PLAN is a detailed description of the requirements for measurements, statistics, inventory, and the records and reports system all quantified as to precision and accuracy, and timeliness. The plan should include information about required programs for scales and balances, calibration, sampling, and similar subjects which are essential to the establishment of performance specifications.

22. MATERIAL CUSTODIAN is an individual assigned responsibility for the control of nuclear material in a localized area of a facility. The localized area should be limited, where practical, to a single material balance area.

23. MEASURED VALUE refers to one or more quantitative or qualitative characteristics that have been determined for a nuclear material item and implies associated limits of error.

    a. The measured value may be quantities of nuclear material determined by sampling and analysis, weight, volume determination, nondestructive assay, or other appropriate means.

    b. The measured value may be calculated on the basis of a chemical analysis or nondestructive assay of a representative sample.

    c. For the purposes of this directive, a discrete, identifiable item is considered to have a measured value if previously measured and if the integrity of the item can be assured according to procedures approved by the cognizant operations office.

24. NONDESTRUCTIVE ASSAY (NDA) is a measurement technique which can provide quantitative or confirmatory measurements of nuclear materials without altering their chemical or physical form.

25. NUCLEAR MATERIALS is a collective term which includes all materials designated from time to time by the Secretary and to which the provisions of this directive apply. (Figure 1 contains a listing of materials currently designated as nuclear materials and includes source material.)

26. NUCLEAR MATERIALS SURVEY refers to the comprehensive examination and evaluation of the effectiveness of the material control and accountability of nuclear materials at DOE contractor facilities.

27. PRECISION is a quantitative measure of the variability of a set of repeated measurements of a given item.

28. PROCESS HOLDUP refers to nuclear material which, although physically inside the process equipment, is part of the flow and is subject to cleanout.

29. PROGRAMMATIC RESPONSIBILITY refers to the function of cognizant Headquarters organizations in sponsoring work in which specified nuclear materials are being used. It includes primary contract administrative responsibility for such programs in instances where no operations office has responsibility for the technical direction of programs.

30. REPORTING IDENTIFICATION SYMBOL (RIS) consists of a unique combination of three or four letters which are assigned to each reporting facility by the Office of Safeguards and Security (DP-30) and/or the Nuclear Regulatory Commission (NRC) for purposes of identifications in the nuclear materials management data base.

31. SAFEGUARDS is an integrated system of physical protection, accountability, and material control measures designed to deter, prevent, detect, and respond to unauthorized possession and use of special nuclear materials. In practice it is the development and application of techniques and procedures dealing with the establishment and continued maintenance of a system of activities including physical protection, quantitative knowledge of the location and use of special nuclear materials, and administrative controls and surveillance to assure that procedures and techniques of the system are effective and are being carried out. Safeguards includes the timely indication of possible diversion or credible assurances by audits and inventory verification that no diversion has occurred.

32. SOURCE MATERIAL refers to (a) uranium, thorium, or any other material determined pursuant to the provisions of Section 61 of the Atomic Energy Act of 1954, as amended, to be source material; or (b) ores containing one or more of the foregoing materials, in such concentration as may by regulation be determined from time to time (see Figure 2).

33. SOURCE AND SPECIAL (SS) MATERIALS (Obsolete term - see nuclear materials).

34. SPECIAL NUCLEAR MATERIAL (SNM) means (a) plutonium, uranium enriched in the isotope 233 or in the isotope 235, and any other material which, pursuant to the provisions of Section 51 of the Atomic Energy Act of 1954, as amended, has been determined to be special nuclear material, but does not include source material; or (b) any material artificially enriched by any of the foregoing, but does not include source material.

35. TAMPER INDICATING DEVICES (TIDs) are devices which may be used on containers and areas which, because of their uniqueness in design or structure, reveal violations of their containment integrity. TIDs include seals, mechanisms, containers, and enclosures.

36. VERIFICATION MEASUREMENT is a quantitative remeasurement to verify an existing measured value as previously recorded.

37. WARNING LIMIT is a quantity limit for inventory differences which when exceeded requires investigation and appropriate action. For processing, production, and fabrication operations, warning limits will be established with a 95 percent probability.

CHAPTER II

REQUIREMENTS

1. GENERAL. The nuclear programs at DOE license-exempt contractors are complex and varied and the safeguards systems (particularly the material control and accountability components of these safeguards systems) are equally technically complex and varied. This Order provides requirements for the material control and accountability components. Facility-specific material control and accountability systems are to be approved at the operations office level.

   a. All nuclear materials (NM) shall be controlled and accounted for as required by this directive. A graded material control and accountability program may be implemented by operations offices using requirements for Category IV (Figure 2) as the minimum for NM, other than SNM consistent with the intrinsic value and operational needs of these materials. A graded material control and accountability program shall be implemented as specified in this directive for special nuclear material.

   b. Each facility shall designate, for approval by the operations office manager, a management official who will be directly responsible for the control of and accounting for nuclear material. This official should be organizationally independent from persons responsible for program operations.

   c. A material control and accountability plan shall be developed for each facility possessing SNM, approved by the cognizant operations office with change control also exercised by that operations office, and submitted to the Office of Safeguards and Security (DP-30) for review. The plan should reflect requirements for containment, surveillance, internal control, measurements, statistics, records and reports system, and inventory certification(s), in the context of quantitatively how precisely is the inventory known at stated risks or confidence levels. It should include such details as programs for scales and balances, calibration and sampling precision constraints, and validation procedures. The elements of this plan should be treated individually and in a collective sense.

   d. Each facility shall have, and require compliance with, one or more current procedure directive(s) implementing its material control and accountability plan covering in specific detail its nuclear material control and accountability activities. These procedures shall be appropriately interfaced with the physical protection and security requirements of the DOE 5632 series to provide the necessary effective integrated safeguards system. Further, these procedures should be developed with consideration of health, safety, and environment. The facility procedure directive(s) must be approved by the cognizant operations office prior to its (or their) implementation.

e. Custodians and handlers of special nuclear material shall receive initial and periodic (at least annual) training on their duties, responsibilities, and obligations.

f. Annually, by December 1, each operations office manager shall submit a report to the Under Secretary, with copies to the Assistant Secretary for Defense Programs, the Inspector General, outlay program managers, and the Director of Safeguards and Security, on the state of safeguards and security of SNM under their responsibility. This report should include, at a minimum:

(1) Improvements achieved in the prior 12 months;

(2) Known deficiencies for which budget funds have been appropriated and the status of corrective action;

(3) Planned improvements which have not yet been placed in the budget cycle;

(4) A summary of the manager's assessment of current capabilities of each facility to meet identified safeguards and security requirements, including residual vulnerabilities, and

(5) Suggested activities that Headquarters could undertake that would assist managers in meeting their responsibilities.

2. ACCOUNTABILITY.

a. Operations offices and facilities shall establish and follow a graded safeguards program for SNM. Graded safeguards is the concept of providing the greatest relative amount of control and effort to that SNM which is most effectively used in a nuclear explosive device. This means that plutonium-239, uranium (≥20% U-235), and uranium-233, in the form of metal and compounds should receive more stringent controls than special nuclear material that must be processed, transmuted, or enriched to make it useable in an explosive device. Material in the most useable form must be inventoried frequently, placed under the tightest administrative controls, and, according to DOE 5632 series requirements, must be subject to the most stringent physical protective measures. A categorization for the graded program for nuclear materials, by material balance area, is shown in Figure 1. Figure 2 shows the minimum requirements for each category. A relative order of attractiveness of specific fabricated material forms and the categorization of these fabricated materials shall be a part of the facility procedure directive(s) which is (are) approved by the cognizant operations office.

The following figure shows the name and reportable quantities of nuclear materials.

Name of Material	Other Nuclear Materials	SNM	Source	Reportable Quantities
Depleted Uranium			X	Kilogram
Enriched Uranium		X		Gram
Plutonium-242		X		Gram
Americium-241	X			Gram
Americium-243	X			Gram
Curium	X			Gram
Berkelium	X			Microgram
Californium	X			Microgram
Plutonium 239-241		X		Gram
Lithium-6	X			Kilogram
Uranium-233		X		Gram
Normal Uranium			X	Kilogram
Neptunium-237	X			Gram
Plutonium-238		X		Gram/tenth
Deuterium	X			Kilogram/tenth
Tritium	X			Gram/hundredth
Thorium			X	Kilogram

Changes to this figure will be made as appropriate.

Figure 1
Reportable Nuclear Materials

Category / Material	IA (Fig. 4 lines 1-11)	IB (Fig. 4 lines 12-24)	II	III	IV
Pu	≥ 2 Kg	≥ 2 Kg	400-1,999g	1-399g	-
U-233	≥ 2 Kg	≥ 2 Kg	400-1,999g	1-399g	-
U-235 (≥ 20%)	≥ 5 Kg	≥ 5 Kg	1-4.999 Kg	1-999g	-
All Other SNM	-	-	-	Reportable Quantities and above	-
Source and Other Nuclear Materials	-	-	-	-	Reportable Quantities and above

Comments on Figure 2:

1. When reference is made to Category I material, both Categories IA and IB are to be included.

2. Refer to DEFINITIONS for quantities or mixtures.

Figure 2

Material Control and Accountability
Categorization of Nuclear Material

Category (See Figure 2)	Inventory Frequency	Measurement & Statistical Control Programs	DOE Survey Frequency
IA	Daily & Bimonthly	Yes	Yearly
IB	Bimonthly	Yes	Yearly
II	Semiannual	Yes	Yearly
III	Annual	Yes	Yearly
IV	Annual	No	Biennial

Comments on Figure 3:

1. A daily inventory is an administrative determination that no gross irregularities appear to exist, no items are obviously missing, and there is no indication that tampering has occurred.

2. Detailed inventory procedures for SNM should be developed to minimize or eliminate unnecessary radiation exposure to personnel. Where the integrity of containment can be assured, daily, bimonthly, and semiannual inventory steps which will necessarily result in radiation exposure to personnel may be extended to annual. Highly radioactive material (>100 rems per hour at one meter from an unshielded surface) is exempt from the inventory requirements of this directive; instead, book records will suffice supplemented by observation, as feasible, and physical containment and administrative controls. Also see page II-12, paragraph 6, Inventory and Control, for additional explanations of permissible practices.

3. Large throughputs, process controls, or other considerations may require operations office managers to prescribe more frequent inventories.

4. Nuclear material surveys should include an evaluation of contractor nuclear material control and accountability practices and performance in complying with DOE requirements.

5. Bimonthly means once every 2 months.

Figure 3

Minimum Graded Nuclear Material Program Requirements

Category IA Material

1. Assembled Plutonium (PU* or U-233) Weapons Components.
2. Assembled U-235 Weapons Components.
3. Pu* or U-233 Machined Weapons Parts.
4. U-235 Machined Weapons Parts.
5. Pu* Metal (buttons, rods, pieces).
6. U-233 Metal.
7. U-235 Metal.
8. Pu* Oxides.
9. U-233 Oxides.
10. U-235 Oxides.
11. U-235 Carbides.

Category IB Material

12. Pu-238 Oxide or Metal.
13. Nitrate Crystals and Nitrate Solutions of Pu, U-233, and U-235.
14. Pu, U-233, and U-235 Solutions other than Nitrate.
15. Compounds of Pu, U-233, and U-235 other than those listed in items 8-12 above.
16. Pu Alloys or Oxide, Carbide or Nitride Mixtures.
17. U-233 Alloys or Oxide, Carbide or Nitride Mixtures.
18. U-235 Alloys or Oxide, Carbide or Nitride Mixtures.
19. Pu Fuel Elements and Assemblies.
20. U-233 Fuel Elements and Assemblies.
21. U-235 Fuel Elements and Assemblies.
22. Pu, U-233, and U-235 High-Grade Recoverable Scrap.
23. Pu, U-233, and U-235 in Irradiated Forms.
24. Pu, U-233, and U-235 Low-Grade Recoverable Scrap (Process Residues).

*Excluding Pu-238 and P-242

Figure 4

Example Ranking of SNM According to Possible Attractiveness for Diversion

b. Each facility shall maintain readily retrievable accountability data by material balance area (MBA) which reflects the quantities of NM which have been received, shipped, or otherwise removed from the MBA and remaining quantities on inventory with entries capable of being updated on a frequency as specified in the following schedule:

(1) For Category IA quantities - daily.

(2) For Category IB quantities and all other NM - at least monthly.

c. Each facility shall establish material balance areas (MBAs) in the sections of the facility where NM are handled and identify and describe such areas and their functions in the facility's material control and accountability directive(s). In general, material balance areas shall meet two criteria:

(1) One individual in each MBA shall be responsible for assuring that all material control and accountability policies are implemented in that MBA; and

(2) Each MBA shall usually conform to the single geographical area concept and be an integral operation. If several geographical areas are to be included in one MBA, all of these areas must be under the administrative control of the same individual and the activity in all of these areas must be associated with an integral operation.

d. If the isotope content of SNM (excluding uranium enriched below 20 percent U-235) transferred between material balance areas is 50 grams or more, the transfer must be measured, or a confirmatory measurement made, by the receiver. Exceptions: Transfers of SNM which (1) are tamper-safed; (2) consist of assembled components with SNM components which are not physically readily accessible; or (3) are sent to laboratories for analysis or examination and return within 72 hours under conditions which provide adequate internal controls to maintain a continuous awareness of the location and integrity of the SNM until it is returned.

e. After obtaining operations office approval for the procedures to be used, each facility shall implement procedures for evaluating SNM inventory differences (IDs). The procedures shall establish control limits, require follow-up investigations when these limits are exceeded, and require statistical tests for trends and bias. Alarm limits and warning limits are to be classified in accordance with CG-S-1, "Classification Guide for Safeguards Information." The local classification officer should be consulted concerning the classification of information used in establishing those limits. The level of investigation required and the assignment of responsibility in the implementation would depend on the material involved and the magnitude of the inventory difference. Similar or modified requirements of this nature at the option of the operations

office manager may be required for other materials listed in the chart of reportable nuclear materials. The procedures for determining the control limits shall be based on one or more of the following: propagation of measurement uncertainties and related variabilities, material flow simulation modeling, or any other approved technique which is based on sound statistical theory and practice. Historical data may be used alone as an interim measure until limits can be established as outlined above.

f. The reporting and investigation of IDs not involving a specific item, which must be in conformance with existing classification guidance, will be as follows:

(1) IDs in Excess of Warning Limits but Less than Alarm Limits.

(a) For Category I and Category II materials, the ID shall be reported promptly to the cognizant operations office manager, and shall be investigated immediately. If the investigations do not result in a satisfactory and compatible explanation within 10 days, the ID shall be reported to the appropriate Headquarters (HQ) program organization, and to the Office of Safeguards and Security, HQ, who will provide copies to other concerned HQ organizations, and through the Assistant Secretary for Defense Programs to the Under Secretary and the Inspector General. If the ID is an underage/"loss" and if the investigation provides a basis for concluding that theft or diversion has taken place, the operations office should promptly notify the local Federal Bureau of Investigation (FBI).

(b) For other SNM, the ID shall be reported promptly to the cognizant operations office and investigated promptly by the contractor.

(2) IDs in Excess of Alarm Limits.

(a) For Category I and Category II materials, the ID shall be reported immediately to the cognizant operations office manager who will report it immediately to the Office of Safeguards and Security, HQ, to the appropriate Headquarters organizations, and to the FBI (in the case of underage/"loss," not overage/gain). The ID shall be investigated immediately. The Office of Safeguards and Security will provide copies of the report to the Under Secretary, the Inspector General, and other concerned HQ organizations.

(b) For other SNM, the ID shall be reported immediately to the cognizant operations office manager. The ID shall be investigated promptly by the contractor.

(3) IDs Involving a Missing Item of SNM.

For any ID involving a missing item of SNM, the missing item shall be investigated immediately and reported to the cognizant operations office at once. The Office of Safeguards and Security (DP-30), the appropriate Headquarters organizations, and the FBI shall be informed immediately of any indication of theft or of possible danger to health and safety of the public, and in any case within 24 hours if the missing item is not found. The Office of Safeguards and Security will provide copies to the Under Secretary (through the Assistant Secretary for Defense Programs), the Inspector General, and other concerned HQ organizations.

3. MEASUREMENT AND STATISTICAL CONTROL PROGRAMS.

a. Each facility shall establish and maintain appropriate measurement program(s) for all SNM. The accuracy and precision of the various techniques shall be consistent with a graded safeguards system. Where required and/or as appropriate, verification or confirmatory measurements may be made using NDA or other accepted measurement methods. A measurement program for other nuclear materials may be conducted where necessary to assure adequate accountability consistent with operational needs and the intrinsic value of these materials.

b. Each facility shall implement measurement control programs for all measurement systems used for the accountability of SNM. These programs shall be consistent with a graded safeguards system and shall be referenceable to and validated by a national measurements and standards program, where practicable, and shall include:

(1) Scale and Balance Program. All scales and balances shall be maintained in good working condition and calibrated pursuant to an established control program.

(2) Analytical Quality Control Program. Data from routine testing shall be analyzed statistically to determine and maintain accuracy and precision of the methods.

(3) Sample Variability Control Program. The uncertainty (variance) associated with each sampling method shall be determined, minimized, and maintained on a current basis.

(4) Control Program for Volume, Temperature, and Pressure Measurements. The precision and accuracy of volume, temperature, and pressure determinations shall be obtained by appropriate techniques, and periodic checks of such calibrations shall be made.

(5) Calibration Program for Nondestructive Assay Measurements. Nondestructive assay instrumentation shall be calibrated with appropriate standards and monitored periodically to ensure proper function within established control limits. Assay predictions periodically should be compared with more accurate measurements of the content of typical materials.

(6) Sample Exchange Program. Facilities should, to the extent practicable, participate in extralaboratory control programs such as the Safeguards Analytical Laboratory Evaluation Program (SALE), to provide external demonstration of the adequacy of an internal quality control program.

c. Measurement control programs for the other NM shall be developed to maintain adequate accountability consistent with the intrinsic and management value of these materials.

4. TAMPER-INDICATING DEVICES.

a. Facilities shall, to the maximum extent practical and possible, use tamper-indicating devices (TIDs) with appropriate written implementing procedures for their use on special nuclear material, unless the cognizant operations office grants an exemption for irradiated items, extruded fuel tubes or targets, or sealed sources. Such devices and procedures must be acceptable to the cognizant DOE operations office. Use of such tamper-indication devices shall include, as appropriate, written procedures for, and compliance with, the following:

(1) Serialization (where appropriate);

(2) Prevention of reusability after the seal has been violated;

(3) Resistance to environmental factors which could conceal any indication of tampering;

(4) Control of their acquisition, shipment, use, and storage;

(5) Retention of comparative samples of TIDs for reference purposes; and

(6) Presence of two authorized individuals when a TID is applied to assure the integrity of the sealed item.

5. EXTERNAL TRANSFERS.

a. The transfers of NM from one facility to another shall be documented by the DOE/NRC Form-741, Nuclear Material Transaction Report, which shall be

prepared and distributed to the principals of the transaction and the cognizant operations office desirably on the day of the transfer, but within 24 hours, or on the first work day after the transfer should it occur on a nonwork day. Measured values should be used for all Category I and II material transfers (see paragraph 5.b.). Measured values should be used for SNM and other NM transfers consistent with the intrinsic and/or monetary value(s) of the NM and as required for associated environmental, safety, and operational controls.

b. All unirradiated Category I and II material transferred between facilities shall have, to the extent possible but consistent with the graded safeguards concept, independently measured values determined by the shipper and the receiver. The shipper's measured values shall be determined prior to shipment unless the integrity of the existing measured values has been assured. Normally, the receiver's measured values shall be determined within 10 calendar days form the date of receipt of the material or transfer document, whichever is later. When this is not possible, confirmatory measurements shall be used on an interim basis for accepting the transfer.

c. As soon as possible, but in no case more than 1 work day after receipt of Category I materials, the identification and integrity of the shipper's tamper-safing devices on each item or container should be verified. The piece count and identification and gross weight of the items or containers received should be checked against the bill of lading, Form DOE/NRC-741/ 741A, or other appropriate shipping document to provide assurance that the shipment was received intact.

   (1) For receipts involving accessible (for sampling or direct NDA measurement) Category I and II materials that are originally accepted by confirmatory receipt measurements, these materials should not enter any chemical separation step until the receiver has independently measured values or unless DOE-approved procedures, including those to evaluate and resolve significant shipper/receiver differences, have been established.

   (2) For receipts involving inaccessible Category I and II material under circumstances which preclude the use of NDA techniques for verification measurements, the shipper's values may be accepted by the receiver without direct or confirmatory measurements if the shipper's TID is found intact by the receiver and an independent agent was present to verify the shipper's package measurement and application of a TID.

d. For shipments of unirradiated SNM greater than 250 grams (multiplying the plutonium and uranium-233 weights by 2.5 and adding to this the uranium-235 weight), and for each discrete item exceeding 250 grams, limits of

error shall be assigned by the shipper and receiver to their measurements, at the 95 percent confidence level, for both the element and isotope values. For shipments of unirradiated accessible SNM of less than 250 grams, the shipper and receiver may estimate the limits of error for the shipment.

e. Calculated limits of error are required for all measurements of external transfers of tritium, except shipments of reservoirs.

f. Limits of error are not required for spent fuel shipments.

g. The requirements of Section 9-50.402 of the DOE Procurement Regulations and other DOE Orders in the 5630 series also apply.

6. INVENTORY AND CONTROL.

a. Each facility having Category IA material shall develop and implement daily administrative control procedures for each MBA having Category IA quantities which take into account the specific nature of the operations within that MBA. These procedures shall be designed to detect abnormal situations and shall be interfaced with the facility's physical protection program. Any noted abnormalities shall be reported immediately to the appropriate facility official.

b. Each facility shall implement a physical inventory program and written inventory plan for nuclear materials which will include:

(1) Bimonthly Physical Inventories.

(a) Facilities shall perform bimonthly physical inventories of Category I materials (Figure 2). All such SNM shall be inventoried with measured values and, where feasible, measurements or estimates of holdup shall be made so that it can be identified in categorizing the inventory difference. Material undergoing processing and recovery operations, and which is inaccessible for measurements by sampling, should be accounted for by use of process data, vessel level and density measurements and calculated concentration values. Plant operational data such as vessel levels, density readings, and sensors should be utilized to monitor process conditions on a routine basis. This process monitoring, in addition to material control procedures, measurements, and specific action criteria, subject to the approval of the cognizant operations office, should be used to routinely track materials in process until operations permit a complete inventory.

(b) The complete inventories should be conducted on a schedule established by the operations office manager, but no less

frequently than annual, unless the facility can demonstrate that an alternative inventory procedure (e.g., dynamic inventory - in-process or continuous) can produce results equivalent to a complete cleanout inventory. The physical inventory procedures shall provide for a documented listing of items showing the serial number, the amount of material, the chemical or physical form, the location, and/or any other data which would be required to uniquely identify and control each item. For inventories containing large numbers of individual items or irradiated nuclear items (e.g., fast critical reactors), statistical sampling plans as approved by the cognizant operations offices may be utilized. Critical facilities require alternative controls to minimize physical handling of the fuel. A system of alternative controls should be clearly defined and approved by the cognizant operations office. Such a system will provide safeguards comparable to that obtained by using frequent physical inventories. The extensive use of inventory sampling plans, tamper-indicating devices, and mechanical inventory techniques, and any other techniques which would minimize personnel exposure to radiation is recommended.

(c) Operations office managers may approve exceptions to inventory frequency requirements for critical assemblies i.e., zero power reactors, when significant time is required to disassemble the critical assembly to gain access to the fuel. However, checks of facility records and inspections of the critical assemblies should be made at the frequency that a physical inventory would otherwise be performed to provide assurance of satisfactory fuel containment. Excepted critical assemblies shall be inventoried during routine disassembly. Irradiated nuclear material in long term storage or in operations which are inaccessible for unique identification or individual piece counting because of radiation levels, should be accounted for based on a principle of containerization where the items are identified, if possible, or (at a minimum) counted at the time the container batch is made up. Containers may be buckets, uniquely identified zones in the fuel storage area, or any other well-defined three dimensional space that will remain unchanged during the inventory period. Containers shall be uniquely identified including serial numbers, dimensions (if applicable), listing of contents, and location or grid number. The control of material in containers should follow the practices for individual items, including the use of such things as TIDs, wherever possible, and control of operations that might lead to unreported changes in the batch content. Further, physical inventories shall be conducted in a systematic manner, shall be designed to avoid duplication, and shall provide for assurance of the integrity of tamper-indicating devices.

(2) Semiannual Physical Inventories. Facilities shall perform semiannual physical inventories of Category II materials (Figure 2). All such materials except in processing and recovery operations shall be, where possible, inventoried using measured values and, where feasible, measurements or estimates of holdup shall be made so that it can be identified in categorizing the inventory difference. Material undergoing processing and recovery operations, and which is inaccessible for measurements by sampling, should be accounted for by use of process data, vessel level and density measurements, and calculated concentration values. Plant operational data such as vessel levels, density readings, and sensors should be utilized to monitor process conditions on a routine basis. This process monitoring, in addition to material control procedures measurements, and specific action criteria, subject to the approval of the cognizant operations office should be used to routinely track materials in process until operations permit a complete inventory. When item identification of individual items cannot be accomplished, there shall be an approved procedure for assuring the accuracy of the accountability records. The complete inventories should be conducted on a schedule established by the operations office manager, but no less frequently than annual, unless the facility can demonstrate that an alternative inventory procedure can produce results equivalent to a complete cleanout inventory. The physical inventories shall be conducted as outlined for bimonthly physical inventories in paragraph 6b(1).

(3) Annual Physical Inventories. Facilities shall perform physical inventories of reportable quantities of all other NM. All such materials in processing and recovery operations shall be inventoried with measured values and, where feasible, measurements or estimates of holdup shall be made so that it can be identified in categorizing the inventory difference. The physical inventories shall be conducted as outlined for bimonthly inventories in paragraph 6b(1).

(4) Special Physical Inventories. Each facility's safeguards program shall include provisions for special inventories of SNM as a result of changes in custodial responsibilities, missing items, large or unexpected inventory differences, or requests from facility management or the manager of the cognizant operations office.

(5) Physical Inventory Reconciliation Program. Each facility shall implement a physical inventory reconciliation program associated with the above physical inventories which is designed to provide assurance that all nuclear material items have been accounted for and that the facility's record system reflects the physical inventory. Upon completion of the physical inventory, the perpetual (book) inventory for each material balance area must be compared with and, if necessary,

be adjusted to the physical inventory. If the material in the inventory is composed of discrete items and a statistical sampling plan is used to conduct the physical inventory, the reconciliation may be made without separately locating and listing the balance of the material in the inventory.

c. Each facility shall implement a physical inventory measurement program for SNM items as follows:

(1) Items With Tamper-Indicating Devices. A confirmatory measurement program shall be developed to ensure the integrity of SNM protected by tamper-indicating devices and may follow the graded concept. These confirmatory measurements may be performed on the basis of a statistical sampling plan during the physical inventory or between inventories.

(2) Items Without Tamper-Indicating Devices. A verification measurement program shall be developed at the MBA level, shall be based on statistical sampling plans, and shall follow the graded concept.

7. INTERNAL CONTROL AND REVIEWS.

a. Operations offices shall assure that their contractors establish and maintain an internal control system. Properly designed and implemented, an internal control system will:

(1) Provide needed information concerning the location or disposition of material quantities;

(2) Reduce the probability of errors in records and reports which might result through improper handling of data or documents, or through use of data of poor quality;

(3) Decrease the risk of and detect theft or other unauthorized or unknown use of material;

(4) Provide a system of checks and balances in the record systems and in the division of duties, so that the work of one person verifies the work of another, and

(5) Provide a system of checks on the materials measurement system.

b. The internal control system shall address:

(1) The organizational structure;

(2) The responsibilities regarding nuclear materials;

(3) The receiving functions,

(4) The shipping functions;

(5) The accountability records system,

(6) The custodial functions including physical inventories,

(7) The materials measurement system;

(8) The effectiveness of the interfacing of nuclear material control and accountability subsystems with each other and the physical protection subsystem of the facility's overall safeguards system, and

(9) The internal audit functions.

c. Each facility possessing Category I quantities of SNM shall provide assurance to the appropriate operations office of the effectiveness of the integrated system of physical protection, and material control and accountability measures in deterring, preventing, detecting, and responding to the theft of SNM.

*U.S. GOVERNMENT PRINTING OFFICE : 1980 0-620-309/187

APPENDIX 2

PRELIMINARY PLANS
FOR CORE DISASSEMBLY AND REMOVAL

APPENDIX 2

PRELIMINARY PLANS FOR CORE DISASSEMBLY AND REMOVAL

While initial plans for disassembly and removal of the TMI-2 core have been formulated,[19] these plans are in the preliminary stage and are constantly being reevaluated[20] by Babcock and Wilcox (B&W) which is being subcontracted by Bechtel to disassemble and remove the core. The current core removal plans do not provide for material accountability measurements or for that matter, very much inspection or measurement (e.g., weighing, neutron counting, gamma ray scanning) except for remote video coverage. In the interests of reducing costs and time, the removal, canning, and transfer of the spent fuel to the storage pool will be done as quickly as possible. Neutron measurements of the fuel assemblies may be difficult to perform because the removal operations will be performed remotely under borated water, and the assemblies will be seal-welded in canisters filled with borated water. The canister will probably be of a size and shape to fit into a shipping cask for a PWR fuel assembly. No plans for inspection of the fuel assemblies in the storage pool area are being made. SNM accountability would most likely have to be performed either in situ in the core or on the extracted fuel assemblies before they are canned. A general description of the preliminary plans (proposed by Bechtel) for core disassembly and removal are given below.

In place core inspection will precede each of the steps in removing the core. For example, after removal of the closure head, cameras may be lowered through the peripheral areas of the plenum to examine part of the top of the core and perform an initial inspection of the periphery of the core. Additional in situ inspection of the fuel assemblies may be performed through the 69 control rod guide tubes. These inspections would provide information on debris or fuel which might have migrated into the plenum assembly and on any mechanical distortions which may have occurred.

Prior to defueling the reactor core, the Incore Detector Assemblies (IDA) may be withdrawn from the fuel assemblies. The ease or difficulty with which they can be removed from the fuel assembly may provide an indication of the extent of damage to the fuel assembly.

For purposes of fuel removal, Bechtel[19] has categorized TMI-2 fuel into three possible damage levels which are described below. The descriptions and general methods for removing the different categories of fuel are excerpted from pages 4-14 to 4-15 in Reference 19.

Damage Level 1

Weakened structurally with pin failures and warpage. Major parts remain intact.

Removal of assemblies with this damage level will be attempted first. A tv camera will be positioned to provide a good view of the upper end fitting. A probe will be inserted into each guide tube in turn and the location of any obstructions noted. Rods with expansion mechanisms on their lower ends will be arranged on a spacer to fit the guide tube pattern and their upper ends will connect to a hoist. These rods will be inserted into the guide tubes to the lower end fitting and the expansion mechanisms will be activated. The lifting forces generated are transmitted to the fuel assembly only at the lower ends of the guide tubes, where the probability of structural integrity is best. The assembly will be lifted out and placed in its storage container after being given a visual examination.

The process will be repeated until removal of all Level 1 assemblies has been accomplished.

Damage Level 2

Significant structural damage and oxidation with fragmented components forming a layer of debris over the intact core portion. No eutectic bonding exhibited.

Visual examination will lead to the conclusion that the assembly has significantly more deterioration than a Level 1 category assembly. Since portions of the assembly may be missing, and since the assembly may be structurally unsound down to perhaps 3 feet from the lower end

fitting, several methods may be necessary to remove the assembly. Rubble and debris present in the upper part of the core, but which is not metallurgically bonded, will be removed with the special tools and vacuum systems to provide clear access to the remaining portions of the fuel assembly.

Removal of the remainder of the assembly may require lifting by a guide tube expansion mechanism and hoist (as described in Damage Level 1 above) or the use of the side-access device, or both, depending on the core location and unique condition of the particular assembly. Any loose pieces falling to the lower vessel cavity upon assembly movement will be retrieved when feasible. Assemblies will be visually examined when removed from the core and placed in storage containers. Side loading cans and side loading tools may be required for assemblies with this degree of damage.

Damage Level 3

Crust of metal oxide eutectic formed which bonds parts of the core together.

Fuel assemblies with this damage level may be mechanically or metallurgically bonded together so that removal intact may be impossible. These fuel assemblies will be separated by cutting them apart. If no fuel assembly can be removed intact, the entire core may have to be cut up with the fragments being removed in storage containers. High speed saws, lasers, or torches may be used to cut or separate these highly damaged assemblies. Special suction equipment and mechanical grappling devices may be used to retrieve debris resulting from the cutting operation.

Before removing the first fuel assembly, a video scan of the top of the core may be performed to ascertain and record the initial core conditions. Once the first fuel assembly has been removed, a camera can be lowered into the vacated location to determine the condition of adjacent fuel assemblies. As the fuel assemblies are lifted out

of the pressure vessel (to be loaded in a can in the pressure vessel and moved) video inspection may be made of the extracted assembly and of the neighboring assemblies in the core.

The removal from the core of dispersed fuel will be carried out by the use of wash/vacuum systems and manipulators. Most likely, the origin of this material with respect to location in the core will be unknown and the opportunities to make some measurement of it or sample it will be very limited. Estimates of the quantities may be feasible by visual access to the dispersed material by remote TV.

M. L. Picklesimer of the Nuclear Regulatory Commission has evaluated the minimum and maximum damage likely in the core.[33] In both cases it was concluded that all fuel rods burst.

APPENDIX 3

ISOTOPIC AND FISSION PRODUCT INVENTORIES FROM PDQ CODE[4]

APPENDIX 3

LEGAL NOTICE

This information is provided by Babcock & Wilcox in connection with a TMI-2 accountability study. Neither Babcock & Wilcox nor any person acting in its behalf:

A. Makes any warranty or representation, express or implied, with respect to the accuracy, completeness, or usefulness of the attached information.

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of, any information provided.

Figure 3-A gives the core location and enrichment of the fuel and lumped burnable poison.

Figure 3-B gives the Fuel 1/4 Core Edit Set 1-52.

Table 3-A gives the average nuclide concentrations at 100 EFPD in atoms/Barn-cm. (3 pages) Note: decimal is at left of number).

Table 3-B gives burnup by fuel assembly in MWD/MTU.

Table 3-C gives burnup at 7 axial locations for each fuel assembly in MWD/MTU.

Table 3-D gives burnup at 32 axial locations for each fuel assembly (number in Table x 10^3 = MWD/MTUH.

To obtain the mass in grams per fuel assembly for each nuclide, multiply the (nuclide concentration per fuel assembly) from Table 3-A by the (gram atomic weight of the nuclide) by (262933.18 atoms Barn/atoms-fuel assembly).

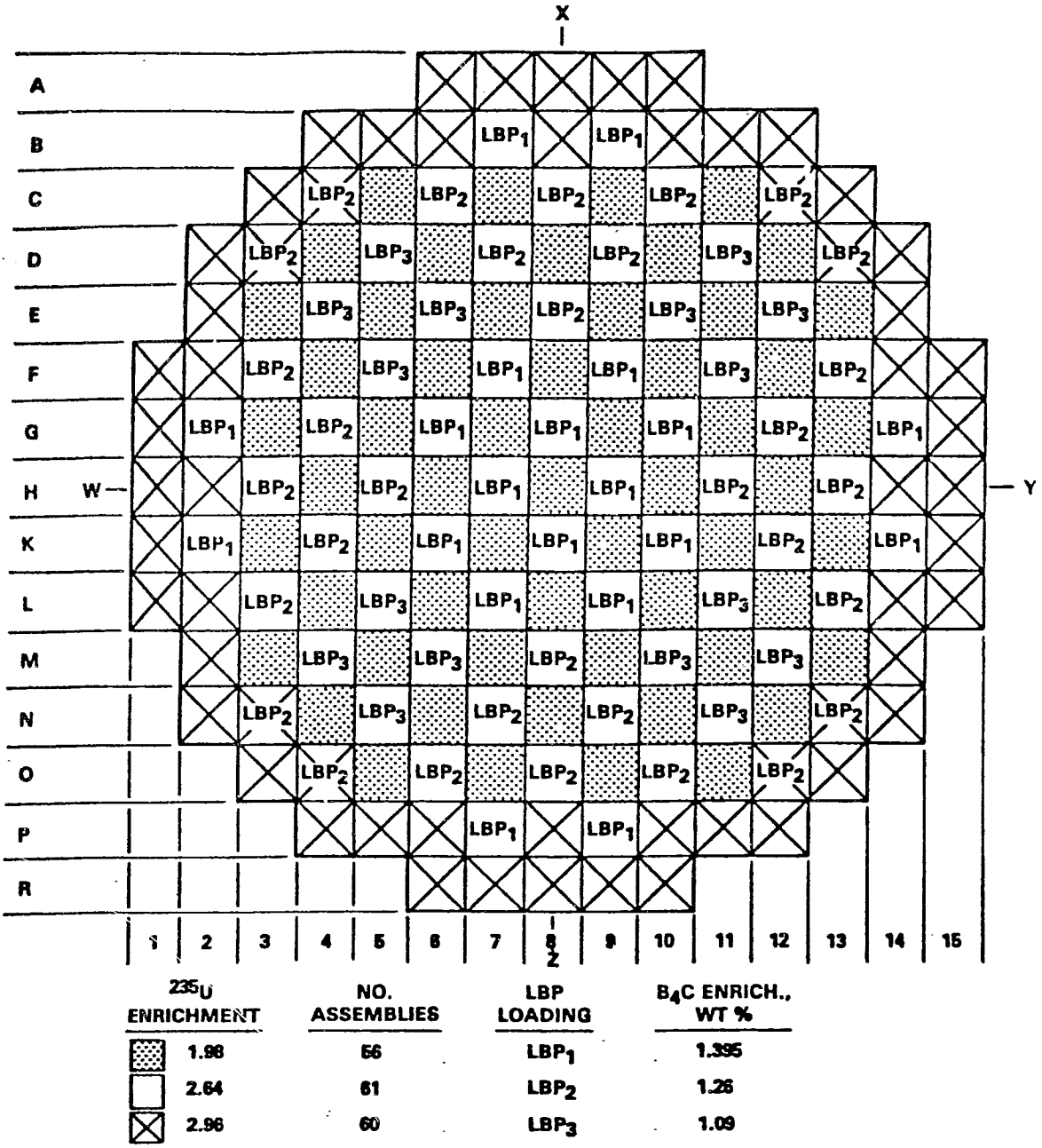

Figure 3-A. Location and Enrichment of Fuel Assemblies and Lumped Burnable Poison in TMI-2 Core.

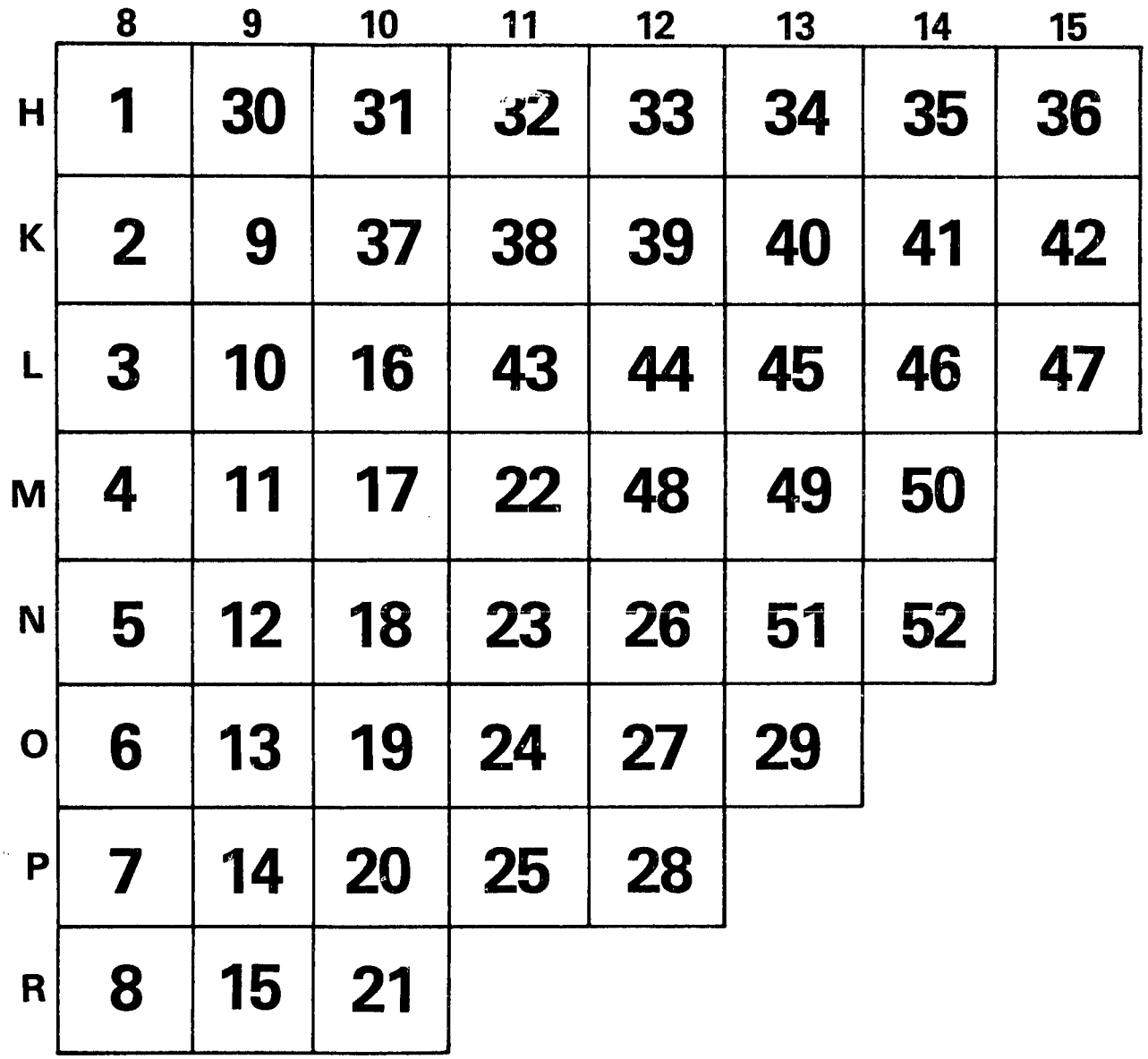

Figure 3-B. TMI-2 Fuel 1/4 Core Edit Set 1-52
(Reference Table 3-A).

AVERAGE NUCLIDE CONCENTRATIONS AT 2400.00 HOURS

EDIT SET	BOR05	LBP10	UFP17	UFP18	PFP19	PFP20	UR235	UR236
1	98500000+03	00000000+00	55215893-04	12964003-04	68456630-05	17826002-05	15956212-03	78736020-05
2	98500000+03	00000000+00	47552236-04	11220806-04	52594021-05	13750033-05	16533106-03	69085897-05
3	98500000+03	00000000+00	41909683-04	98669148-05	59553851-05	15540897-05	11980983-03	57303933-05
4	98500000+03	00000000+00	42565429-04	10084145-04	41427866-05	10863274-05	16900804-03	61748510-05
5	98500000+03	00000000+00	36825377-04	87081317-05	46628008-05	12209137-05	12318356-03	50668273-05
6	98500000+03	00000000+00	43395651-04	10273414-04	43442003-05	11385596-05	16840344-03	63076181-05
7	98500000+03	00000000+00	52313895-04	12350171-04	51648291-05	13505181-05	18576494-03	76023158-05
8	98500000+03	00000000+00	37024636-04	88224295-05	26175528-05	68877228-06	19719791-03	53630127-05
9	98500000+03	00000000+00	43327880-04	10186429-04	63914799-05	16661022-05	11878293-03	59224112-05
10	98500000+03	00000000+00	44110051-04	10437322-04	44729498-05	11718225-05	16786810-03	64021956-05
11	98500000+03	00000000+00	39238505-04	92619031-05	51873549-05	13563201-05	12174849-03	53689982-05
12	98500000+03	00000000+00	40166975-04	95329931-05	36767220-05	96543129-06	17078648-03	58307295-05
13	98500000+03	00000000+00	39294555-04	92744449-05	52139924-05	13632367-05	12171058-03	53782297-05
14	98500000+03	00000000+00	41157721-04	97579212-05	39459290-05	10353514-05	17007989-03	60097042-05
15	98500000+03	00000000+00	34146290-04	81522709-05	22151993-05	58380218-06	19935736-03	49478085-05
16	98500000+03	00000000+00	40032854-04	94422715-05	54084135-05	14133129-05	12117131-03	54763693-05
17	98500000+03	00000000+00	40470006-04	96030772-05	37244189-05	97779714-06	17055733-03	58682943-05
18	98500000+03	00000000+00	32717077-04	77659973-05	36642980-05	96230514-06	12656086-03	45272807-05
19	98500000+03	00000000+00	38955016-04	92529336-05	34728744-05	91249230-06	17169320-03	56665832-05
20	98500000+03	00000000+00	44087781-04	10465852-04	36258[illegible]18-05	95198171-06	19191275-03	64067050-05
21	98500000+03	00000000+00	26171651-04	62770564-05	1325[illegible]292-05	35032735-06	20534107-03	37769048-05
22	98500000+03	00000000+00	36402834-04	86156444-05	44359690-05	11622333-05	12380923-03	49847234-05
23	98500000+03	00000000+00	36960384-04	87933347-05	30843404-05	81138885-06	17315821-03	53623888-05
24	98500000+03	00000000+00	33450218-04	79378764-05	37352411-05	98062113-06	12596419-03	45862748-05
25	98500000+03	00000000+00	33875241-04	80922399-05	21285185-05	56130984-06	19955207-03	48949230-05
26	985000[illegible]+03	00000000+00	32437007-04	77043341-05	35080700-05	92159970-06	12670394-03	44498002-05
27	985000[illegible]+03	00000000+00	30734536-04	73538085-05	18289070-05	48310612-06	20196338-03	45588719-05
28	98500000+03	00000000+00	21141058-04	50872352-05	85522177-06	22673180-06	20912649-03	30530968-05
29	98000000+03	00000000+00	21962272-04	52833583-05	90824700-06	24077244-06	20850818-03	31735952-05
30	98500000+03	00000000+00	47552340-04	11220830-04	52594257-05	13750094-05	16533099-03	69086041-05
31	98500000+03	00000000+00	41909929-04	98669728-05	59554575-05	15541085-05	11980966-03	57304270-05
32	98500000+03	00000000+00	42565862-04	10084243-04	41428706-05	10863491-05	16900775-03	61749115-05
33	98500000+03	00000000+00	36825832-04	87082354-05	46629293-05	12209446-05	12318323-03	50668885-05
34	98500000+03	00000000+00	43396243-04	10273548-04	43443216-05	11385910-05	16840303-03	63077010-05
35	98500000+03	00000000+00	52314573-04	12350324-04	51649[illegible]1-05	13505542-05	18576446-03	76024128-05
36	98500000+03	00000000+00	37025098-04	88225376-05	26176241-05	68879073-06	19719754-03	53630799-05
37	98500000+03	00000000+00	44110220-04	10437362-04	44729843-05	11718315-05	16786799-03	64022202-05
38	98500000+03	00000000+00	39238779-04	92619655-05	51874306-05	13563396-05	12174828-03	53690348-05
39	98500000+03	00000000+00	40167411-04	95330938-05	36768043-05	96545264-06	17078620-03	58307922-05
40	98500000+03	00000000+00	39295049-04	92745580-05	52141313-05	13632725-05	12171023-03	53782956-05
41	98500000+03	00000000+00	41158331-04	97580618-05	39460521-05	10353833-05	17007940-03	60097913-05
42	98500000+03	00000000+00	34146793-04	81523885-05	22152685-05	58382026-06	19935697-03	49478806-05
43	98500000+03	00000000+00	40470148-04	96031111-05	37244460-05	97780411-06	17055724-03	58683148-05
44	98500000+03	00000000+00	32717308-04	77660505-05	36643536-05	96231963-06	12656066-03	45273124-05
45	98500000+03	00000000+00	38955449-04	92530332-05	34729552-05	91251322-06	17169288-03	56666446-05
46	98500000+03	00000000+00	44088369-04	10465987-04	36259035-05	95200815-06	19191230-03	64067899-05
47	98500000+03	00000000+00	26172024-04	62771445-05	13250695-05	35033792-06	20534076-03	37769588-05

TABLE 3-A

AVERAGE NUCLIDE CONCENTRATIONS AT 100 EFPD

AVERAGE NUCLIDE CONCENTRATIONS AT 2400.00 HOURS

EDIT SET	BOROS	LBP10	UFP17	UFP18	PFP19	PFP20	UR235	UR236
48	98500000+03	00000000+00	36960518-04	87933661-05	30843645-05	81139520-06	17315809-03	53624089-05
49	98500000+03	00000000+00	33450470-04	79379340-05	37353001-05	98063637-06	12596402-03	45863089-05
50	98500000+03	00000000+00	33875614-04	80923269-05	21285669-05	56132251-06	19955182-03	48949767-05
51	98500000+03	00000000+00	30734663-04	73538383-05	18289223-05	48311017-06	20196330-03	45588903-05
52	98500000+03	00000000+00	21141221-04	50872736-05	85523531-06	22673536-06	20912638-03	30531197-05
53	98500000+03	00000000+00	37128015-04	88111447-05	35892758-05	94129366-06	16547618-03	52973524-05

Table 3-A Continued

AVERAGE NUCLIDE CONCENTRATIONS AT 2400.00 HOURS

EDIT SET	NP237	UR238	PU239	PU240	PU241	PU242	AM243	I0053
1	14357570-06	72814093-02	17988640-04	18866274-05	55670244-06	26739595-07	65411851-09	99557467-08
2	11476948-06	72844345-02	16887456-04	15265060-05	37972083-06	14950028-07	32673734-09	86825081-08
3	91072066-07	73349941-02	16024449-04	16840264-05	43278898-06	20576635-07	43432401-09	79142342-08
4	88916154-07	72877942-02	15109774-04	12369325-05	27573825-06	95559461-08	17934497-09	78078800-08
5	70981071-07	73327897-02	14611432-04	13667962-05	30965340-06	12617677-07	23137646-09	70016027-08
6	93834395-07	72870963-02	15526480-04	12922445-05	29232932-06	10317813-07	20054468-09	77234807-08
7	12021512-06	72609210-02	16337665-04	14812042-05	39446148-06	15622696-07	32736331-09	88358161-08
8	59117281-07	72623284-02	11644582-04	80351057-06	16231100-06	48636967-08	77517848-10	60311909-08
9	98205444-07	73339323-02	16514438-04	17889913-05	47532683-06	23546615-07	52046797-09	81709007-08
10	96493905-07	72867695-02	15661689-04	13237202-05	30584717-06	11050770-07	21772318-09	80789969-08
11	78652382-07	73369317-02	15106923-04	14950786-05	35961372-06	15793743-07	30558906-09	73950772-08
12	78567654-07	72892296-02	14344121-04	11133953-05	23364164-06	75700764-08	13233857-09	72805357-08
13	79055928-07	73368582-02	15167244-04	15027834-05	36118738-06	15884906-07	30829165-09	71749215-08
14	85816395-07	72881224-02	15049673-04	11902254-05	25390811-06	84224886-08	15643051-09	71076648-08
15	49797339-07	72707702-02	10798735-04	69180912-06	12854852-06	35121699-08	50776208-10	55727833-08
16	82201769-07	73363640-02	15378339-04	15500420-05	38040781-06	17107579-07	33961380-09	75529013-08
17	79419770-07	72891285-02	14380836-04	11253231-05	23846890-06	78083636-08	13742258-09	73836325-08
18	56569462-07	73405958-02	13423157-04	11108519-05	21777352-06	75751788-08	12263339-09	60434617-08
19	74392980-07	72898082-02	14061992-04	10599551-05	21464630-06	66956921-08	11377418-09	68714107-08
20	83118093-07	72655118-02	13912167-04	10859924-05	24535340-06	80774279-08	13902027-09	74187477-08
21	29142587-07	72747677-02	82352542-05	42881261-06	64921837-07	14558230-08	17174121-10	42493754-08
22	66624134-07	73389333-02	14122955-04	13048034-05	29057854-06	11668807-07	20459633-09	67973984-08
23	65264376-07	72912209-02	13227351-04	95170790-06	18310003-06	53859309-08	84340686-10	66350126-08
24	55638898-07	73409033-02	13130943-04	11219216-05	22888329-06	83483466-08	13229426-09	60292282-08
25	47268070-07	72710982-02	10596226-04	66806708-06	12109796-06	31911796-08	43113018-10	56629372-08
26	52160505-07	73415609-02	12776931-04	10610181-05	20989839-06	73943030-08	25544222-10	53174655-08
27	43448228-07	72713158-02	10649578-04	59047510-06	91352946-07	20311202-08	25544222-10	53174655-08
28	18671272-07	72770491-02	67177994-05	28565110-06	34493981-07	61147533-09	56708505-11	35311676-08
29	19868564-07	72766752-02	69850862-05	30310883-06	36896809-07	65088698-09	59636877-11	37269429-08
30	11476997-06	72844346-02	16887489-04	15265121-05	37972311-06	14950156-07	32674087-09	86825008-08
31	91073205-07	73349934-02	16024528-04	16840441-05	43279606-06	20577122-07	43433734-09	79142535-08
32	88917955-07	72877921-02	15109899-04	12369545-05	27574598-06	95563252-08	17935406-09	78079319-08
33	70982902-07	73317895-02	14611584-04	13668265-05	30966424-06	12618305-07	23139134-09	70016638-08
34	93837014-07	72870955-02	15526660-04	12922759-05	29234041-06	10318366-07	20006854-09	77235286-08
35	12021846-06	72609202-02	16337854-04	14812388-05	39447548-06	15623481-07	32738492-09	88358405-08
36	59118881-07	72693284-02	11644723-04	80353007-06	16231701-06	48639479-08	77523045-10	60311960-08
37	96494664-07	72867693-02	15661740-04	13237291-05	30585040-06	11050936-07	21772739-09	80790125-08
38	78653550-07	73369316-02	15107013-04	14950974-05	35962081-06	15794189-07	30560031-09	73951098-08
39	78569451-07	72892294-02	14344261-04	11134172-05	23364881-06	75704026-08	13234599-09	72805834-08
40	79058086-07	73368576-02	15167412-04	15028180-05	36120026-06	15885712-07	30831219-09	71749666-08
41	85819080-07	72881217-02	15049876-04	11902581-05	25391889-06	84229890-08	15644253-09	71077115-08
42	49798880-07	72707697-02	10798886-04	69182858-06	12855410-06	35123819-08	50780179-10	55728087-08
43	79420361-07	72891284-02	14380882-04	11253303-05	23847128-06	78084722-08	13742501-09	73836492-08
44	56570337-07	73405953-02	13423245-04	11108669-05	21777810-06	75754025-08	12263810-09	60434876-08
45	74394728-07	72898079-02	14062135-04	10599770-05	21465316-06	66959915-08	11378081-09	68714496-08
46	83120452-07	72655114-02	13912337-04	10860191-05	24536271-06	80778581-08	13902989-09	74187923-08
47	29143466-07	72747678-02	82353695-05	42882460-06	64924630-07	14559091-08	17175426-10	42493990-08

Table 3-A Continued

AVERAGE NUCLIDE CONCENTRATIONS AT 2400.00 HOURS

EDIT SET	NP237	UR238	PU239	PU240	PU241	PU242	AM243	I0053
48	65264902-07	72912209-02	13227397-04	95171463-06	18310203-06	53860117-08	84342339-10	66350287-08
49	55639806-07	73409032-02	13131032-04	11219372-05	22888823-06	83485999-08	13229953-09	60292558-08
50	47269157-07	72710979-02	10596334-04	66808082-06	12110180-06	31913189-08	43115448-10	56629722-08
51	43448598-07	72713159-02	10649618-04	59047967-06	91354039-07	20311535-08	25544759-10	53174802-08
52	18671563-07	72770490-02	67178490-05	28585532-06	34494768-07	61149435-09	56710752-11	35311862-08
53	68621938-07	72981996-02	13228164-04	10703441-05	23378711-06	86413430-08	16064085-09	66025997-08

Table 3-A Continued

AVERAGE NUCLIDE CONCENTRATIONS AT 2400.00 HOURS

EDIT SET	XE054	PM061	SM062	GAD	BURNUP	BURNUP	NP239	ENRPU
1	26489981-08	24318770-07	41213547-07	00000000+00	00000000+00	51071256+04	87575465-06	00000000+00
2	26343171-08	20899573-07	41776746-07	00000000+00	00000000+00	43396707+04	79356702-06	00000000+00
3	20054890-08	19672501-07	31531437-07	00000000+00	00000000+00	39361946+04	79007622-06	00000000+00
4	25460127-08	18515462-07	40705159-07	00000000+00	00000000+00	38350471+04	70214319-06	00000000+00
5	19460127-08	18515462-07	31163091-07	19230769-01	00000000+00	34094794+04	70218196-06	00000000+00
6	25504595-08	18376802-07	41037746-07	00000000+00	00000000+00	39202593+04	69831851-06	00000000+00
7	28227340-08	21003127-07	44880064-07	00000000+00	00000000+00	47240040+04	72298543-06	00000000+00
8	24822027-08	13857334-07	42300841-07	00000000+00	00000000+00	32514987+04	47135628-06	00000000+00
9	20231821-08	20417709-07	31789690-07	00000000+00	00000000+00	40899028+04	81823548-06	00000000+00
10	25742808-08	19247979-07	41043506-07	00000000+00	00000000+00	39899771+04	73030506-06	00000000+00
11	19696471-08	18201731-07	31060504-07	00000000+00	00000000+00	36513596+04	73431275-06	00000000+00
12	24973876-08	17150292-07	40329670-07	00000000+00	00000000+00	35984507+04	65195331-06	00000000+00
13	19599943-08	17673377-07	31111847-07	00000000+00	00000000+00	36582101+04	71195424-06	00000000+00
14	25079095-08	16833547-07	41060188-07	00000000+00	00000000+00	37023805+04	64710256-06	00000000+00
15	24175637-08	12707890-07	41902056-07	00000000+00	00000000+00	29810630+04	43305230-06	00000000+00
16	19805882-08	18644445-07	31197080-07	00000000+00	00000000+00	37354196+04	75108872-06	00000000+00
17	25023922-08	17402818-07	40277339-07	00000000+00	00000000+00	36274371+04	65942195-06	00000000+00
18	19019508-08	14565464-07	30708561-07	00000000+00	00000000+00	29861876+04	61432915-06	00000000+00
19	24648320-08	16143688-07	40298072-07	00000000+00	00000000+00	34815355+04	61639360-06	00000000+00
20	26805177-08	17293446-07	43647936-07	00000000+00	00000000+00	39166147+04	59495910-06	00000000+00
21	21560654-08	95041279-08	40271651-07	00000000+00	00000000+00	22520091+04	32004733-06	00000000+00
22	19262516-08	16554486-07	30560013-07	00000000+00	00000000+00	33547074+04	67084742-06	00000000+00
23	24257871-08	15480724-07	39687753-07	00000000+00	00000000+00	32851989+04	58809742-06	00000000+00
24	18690283-08	14529606-07	30104469-07	00000000+00	00000000+00	30528439+04	59182824-06	00000000+00
25	24280951-08	12877215-07	41642792-07	00000000+00	00000000+00	29514681+04	43574627-06	00000000+00
26	18588140-08	14223782-07	29937949-07	00000000+00	00000000+00	29505339+04	58153097-06	00000000+00
27	24590880-08	12035803-07	43192679-07	00000000+00	00000000+00	29505339+04	44222679-06	00000000+00
28	20002165-08	77830248-07	39335525-07	00000000+00	00000000+00	26680910+04	44222679-06	00000000+00
29	20559909-08	82250386-08	39568762-07	00000000+00	00000000+00	18001427+04	26347463-06	00000000+00
30	26343168-08	20899562-07	41776763-07	00000000+00	00000000+00	43396814+04	79356618-06	00000000+00
31	20054906-08	19672567-07	31531476-07	00000000+00	00000000+00	39362218+04	79007798-06	00000000+00
32	25460173-08	18515606-07	40705212-07	00000000+00	00000000+00	38350900+04	70214772-06	00000000+00
33	19599880-08	17126259-07	31163150-07	19230769-01	00000000+00	34095268+04	70218804-06	00000000+00
34	25504646-08	18376946-07	41037829-07	00000000+00	00000000+00	39203086+04	69832291-06	00000000+00
35	28227374-08	21003219-07	44880154-07	00000000+00	00000000+00	47240718+04	72298813-06	00000000+00
36	24822041-08	13857364-07	42300906-07	00000000+00	00000000+00	32515432+04	47135705-06	00000000+00
37	25742822-08	19248024-07	41043529-07	00000000+00	00000000+00	39899942+04	73030643-06	00000000+00
38	19696494-08	18201829-07	31060543-07	00000000+00	00000000+00	36513886+04	73431616-06	00000000+00
39	24973931-08	17150426-07	40329743-07	00000000+00	00000000+00	35984936+04	65195783-06	00000000+00
40	19599985-08	17673521-07	31111930-07	00000000+00	00000000+00	36582624+04	71195907-06	00000000+00
41	25079151-08	16833688-07	41060282-07	00000000+00	00000000+00	37024414+04	64710723-06	00000000+00
42	24175678-08	12707966-07	41902124-07	00000000+00	00000000+00	29811102+04	43305459-06	00000000+00
43	25023941-08	17402864-07	40277362-07	00000000+00	00000000+00	36274514+04	65942354-06	00000000+00
44	19019533-08	14565540-07	30708559-07	00000000+00	00000000+00	29862114+04	61433206-06	00000000+00
45	24648368-08	16143800-07	40298143-07	00000000+00	00000000+00	34815781+04	61639743-06	00000000+00
46	26805230-08	17293575-07	43648013-07	00000000+00	00000000+00	39166716+04	59496322-06	00000000+00
47	21560703-08	95041913-07	40271707-07	00000000+00	00000000+00	22520433+04	32004933-06	00000000+00

Table 3-A Continued

AVERAGE NUCLIDE CONCENTRATIONS AT 2400.00 HOURS

EDIT SET	XE054	PM061	SM062	GAD	BURNUP	BURNUP	NP239	ENRPU
48	24257890-08	15480768-07	39687772-07	00000000+00	00000000+00	32852124+04	58809901-06	00000000+00
49	18690313-08	14529687-07	30104515-07	00000000+00	00000000+00	30528693+04	59183114-06	00000000+00
50	24281007-08	12877307-07	41642845-07	00000000+00	00000000+00	29515032+04	43574917-06	00000000+00
51	24590907-08	12035841-07	43192699-07	00000000+00	00000000+00	26681026+04	44222810-06	00000000+00
52	20002212-08	77830694-07	35335556-07	00000000+00	00000000+00	18001571+04	26347608-06	00000000+00
53	22856013-08	15648795-07	37923211-07	00000000+00	00000000+00	33424164+04	59369150-06	00000000+00

Table 3-A Continued

**** FULL CERTIFICATION **** **** FULL CERTIFICATION **** **** FULL CERTIFICATION **** TM2FLKB
(FLAM3 V2.1 CREATED 12/26/78) - TM2FLKB - 03/29/79 CASE 10 (8,8,32) TMI-2 FLAM 68.6- 79.6EFPD 293.5 PWD APSR PAGE 629

EXPOSURE (MWD/MTU)

I/J	8	9	10	11	12	13	14	15
H	3912.75	3373.68	3129.11	3003.88	2780.47	3075.16	3590.79	2543.18
K	3373.68	3270.55	3104.97	2994.82	2831.25	2947.77	2947.93	2344.71
L	3129.11	3104.97	3030.81	2877.92	2476.65	2768.68	3089.06	1754.51
M	3003.88	2994.82	2877.92	2734.01	2602.04	2498.08	2323.74	
N	2780.47	2831.25	2476.65	2602.04	2422.92	2147.43	1451.95	
O	3075.16	2947.77	2768.68	2498.08	2147.43	1537.74		
P	3590.79	2947.93	3089.06	2323.74	1451.95			
R	2543.18	2344.71	1754.51					

TABLE 3-B

BURNUP BY FUEL ASSEMBLY IN MWD/MTUH

TABLE 3-C

BURNUP BY FUEL ASSEMBLY AT 7 AXIAL LOCATIONS IN MWD/MTUH

EXPOSURE (MWD/MTU)

I/J	8	9	10	11	12	13	14	15
	1428.09	1494.40	1223.93	1489.20	1418.56	1458.57	1293.81	1049.09
	3812.79	3361.20	3199.27	3130.45	2960.82	3119.74	3460.80	2461.65
	4963.17	4277.66	4071.95	3874.95	3633.10	3913.47	4515.59	3170.11
H	5163.10	4384.17	4086.37	3796.89	3472.73	3898.14	4684.69	3312.28
	5011.28	4204.37	3871.17	3554.82	3213.18	3708.56	4589.60	3251.86
	4456.34	2178.91	1990.44	1935.71	1758.48	2022.56	2398.29	1626.46
	2554.34	2178.91	1990.44	1935.71	1758.48	2022.56	2398.29	1626.46
	1494.40	1527.87	1510.70	1528.12	1526.38	1440.58	1276.85	1008.50
	3361.20	3318.72	3199.84	3193.48	3082.25	3070.56	2906.99	2289.50
	4277.66	4170.46	3983.11	3910.56	3758.50	3806.97	3717.08	2929.40
K	4384.17	4227.69	3954.16	3736.95	3416.93	3694.94	3786.91	3041.81
	4204.37	4025.80	3726.05	3454.60	3076.81	3466.04	3658.37	2972.78
	3715.03	3575.08	3362.18	3235.21	3065.43	3251.64	3331.66	2678.77
	2178.91	2048.27	1998.27	1904.83	1892.43	1903.64	1957.66	1492.19
	1223.93	1510.70	1531.43	1509.97	1495.38	1397.10	1151.76	731.95
	3199.27	3199.84	3230.21	3170.28	3114.73	2965.06	3074.23	1741.81
	4071.95	3983.11	3962.74	3866.40	3738.12	3662.62	3961.04	2224.08
L	4086.37	3954.16	3787.08	3463.83	2533.90	3361.69	3983.35	2276.36
	3871.17	3726.05	3499.42	3087.65	1844.82	3060.11	3835.37	2204.90
	3460.66	3362.18	3275.56	3114.25	2779.74	3057.27	3570.17	1996.28
	1990.44	1998.72	1929.72	1933.05	1829.84	1876.91	2047.45	1106.21
	1489.20	1528.12	1509.97	1170.16	1377.74	1266.61	1052.97	
	3130.45	3193.48	3170.28	3018.19	2894.60	2670.86	2374.90	
	3874.95	3910.56	3866.40	3727.38	3516.99	3261.94	2970.71	
M	3796.89	3736.95	3463.83	3373.20	3098.89	3092.63	2960.15	
	3554.82	3454.60	3087.65	3029.38	2742.26	2865.24	2826.76	
	3245.17	3235.21	3114.25	3004.89	2822.71	2730.74	2607.64	
	1935.71	1904.83	1933.83	1814.90	1761.13	1598.55	1473.08	
	1418.56	1526.38	1495.38	1377.74	1279.74	1133.50	715.97	
	2960.82	3082.25	3114.73	2894.60	2635.76	2263.60	1512.62	
	3633.10	3758.50	3738.12	3516.99	3184.52	2755.01	1857.33	
N	3472.73	3416.93	2533.90	3098.89	2971.08	2673.69	1842.73	
	3213.18	3076.81	1844.82	2742.26	2722.37	2503.76	1748.46	
	3006.40	3065.43	2779.74	2822.71	2615.68	2319.73	1591.58	
	1758.48	1892.43	1829.84	1761.13	1551.53	1382.50	895.12	

Table 3-C Continued

**** FULL CERTIFICATION **** **** FULL CERTIFICATION **** **** FULL CERTIFICATION **** TM2FLKB
(FLAM3 V2.1 CREATED 12/26/78) - TM2FLKB - 03/29/79 CASE 10 (8,8,32) TMI-2 FLAM 68.6- 79.6EFPD 293.5 PWD APSR PAGE 628

EXPOSURE (MWD/MTU)

I/J	8	9	10	11	12	13	14	15
	1458.57	1440.58	1397.10	1266.61	1133.89	803.53		
	3119.74	3070.56	2965.06	2670.86	2263.40	1623.91		
	3913.47	3806.97	3662.62	3261.94	2755.01	1967.76		
O	3898.14	3694.94	3361.69	3092.63	2673.69	1933.15		
	3708.56	3466.04	3060.11	2865.24	2503.76	1821.73		
	3405.11	3251.64	3057.24	2730.74	2319.73	1662.35		
	2022.56	1903.64	1876.91	1598.55	1382.50	951.74		
	1293.81	1276.85	1151.76	1052.97	715.97			
	3460.80	2906.99	2074.23	2374.90	1512.42			
	4515.59	3717.08	3961.04	2970.71	1857.33			
P	4684.69	3786.91	3983.35	2960.15	1842.73			
	4589.60	3658.67	3835.37	2826.76	1748.46			
	4192.76	3331.66	3570.19	2607.64	1591.58			
	2398.29	1957.66	2047.45	1473.08	895.12			
	1049.09	1008.50	731.95					
	2461.65	2289.50	1741.81					
	3170.11	2929.40	2224.08					
R	3312.28	3041.81	2276.36					
	3251.86	2972.78	2204.90					
	2930.77	2678.77	1996.28					
	1626.46	1492.19	1106.21					

TABLE 3-D

BURNUP BY FUEL ASSEMBLY AT 32 AXIAL LOCATIONS

Number in Table x 10^3 = MWD/MTUH

CORE AVERAGE EXPOSURE = 2.66200E+00

EXPOSURE BY CHANNEL

I/J	1	2	3	4	5	6	7	8
1	3.91275	3.37368	3.12911	3.00388	2.78047	3.07516	3.59079	2.54318
2	3.37368	3.27055	3.10497	2.99482	2.83125	2.94777	2.74793	2.34471
3	3.12911	3.10497	3.03081	2.87792	2.47665	2.76868	3.08906	1.75451
4	3.00388	2.99482	2.87792	2.73401	2.60204	2.49808	2.32374	
5	2.78047	2.83125	2.47665	2.60204	2.42292	2.14743	1.45195	
6	3.07516	2.94777	2.76868	2.49808	2.14743	1.53774		
7	3.59079	2.94793	3.08906	2.32374	1.45195			
8	2.54318	2.34471	1.75451					

EXPOSURE BY BANK

(I = 1)

K/J	1	2	3	4	5	6	7	8
32	.46235	.61702	.40270	.64333	.58964	.62304	.41602	.38577
31	.96889	1.17736	.83939	1.20758	1.11209	1.17440	.87286	.76366
30	1.55467	1.59123	1.34077	1.59527	1.54221	1.55925	1.40648	1.13137
29	2.06199	2.03160	1.75953	1.99785	1.93148	1.96346	1.87265	1.47902
28	2.59092	2.47503	2.19227	2.38660	2.29153	2.35724	2.35648	1.81310
27	3.26556	2.92035	2.76420	2.75884	2.62177	2.73554	2.96341	2.13567
26	3.76673	3.30573	3.17021	3.08184	2.91289	3.06697	3.41646	2.41698
25	4.15571	3.62239	3.47855	3.34992	3.15944	3.34466	3.77057	2.65211
24	4.46044	3.87483	3.71609	3.56457	3.35775	3.56944	4.05047	2.84354
23	4.69734	4.07020	3.89660	3.72837	3.5069	3.74377	4.26982	2.99546
22	4.87708	4.21560	4.02753	3.84454	3.61399	3.87109	4.43705	3.11250
21	5.00811	4.31721	4.11251	3.91592	3.67493	3.95441	4.55871	3.19887
20	5.09744	4.38117	4.15722	3.94557	3.69229	3.99691	4.63985	3.25872
19	5.15181	4.41319	4.16838	3.93721	3.66775	4.00231	4.66614	3.29601
18	5.17758	4.41949	4.15220	3.89713	3.60572	3.97670	4.70360	3.31511
17	5.18089	4.40628	4.11641	3.83518	3.51820	3.92954	4.70024	3.32043
16	5.16790	4.37980	4.06922	3.76475	3.42431	3.87330	4.68421	3.31647
15	5.14386	4.34518	4.01810	3.69631	3.34232	3.81905	4.66287	3.30692
14	5.11196	4.30594	3.96772	3.63763	3.27890	3.77292	4.64072	3.29429
13	5.07328	4.26326	3.92023	3.59000	3.23377	3.73626	4.61928	3.27914
12	5.02630	4.21587	3.87616	3.55217	3.20399	3.70743	4.59684	3.26016
11	4.96699	4.15991	3.83017	3.52050	3.18663	3.68255	4.56905	3.23410
10	4.88918	4.08950	3.77544	3.77544	3.48788	3.17607	3.65425	4.52790
9	4.78319	3.99627	3.70237	3.44250	3.15944	3.61047	4.46136	3.13777
8	4.63748	3.86985	3.59893	3.36816	3.11542	3.53467	4.35477	3.05115
7	4.43895	3.69891	3.45211	3.24809	3.02132	3.40958	4.19169	2.92557
6	4.17387	3.47172	3.24979	3.06886	2.86393	3.22111	3.95586	2.74990
5	3.82763	3.17803	2.98109	2.82147	2.63595	2.95941	3.63153	2.51269
4	3.38529	2.81020	2.63622	2.50074	2.33280	2.61882	3.20392	2.20175
3	2.82825	2.36765	2.20326	2.10726	1.94735	2.20113	2.65711	1.80324
2	2.12039	1.91001	1.65455	1.69366	1.45913	1.76339	1.96596	1.30268
1	1.15585	1.05685	.90163	.93503	.79324	.97156	1.06147	.69176

EXPOSURE BY BANK

(I = 2)

K/J	1	2	3	4	5	6	7	8
32	.61702	.61933	.65181	.63294	.67861	.58973	.52115	.38503
31	1.17736	1.17826	1.22385	1.19553	1.26007	1.11833	.99975	.75048
30	1.59123	1.65364	1.61559	1.66153	1.63841	1.56268	1.35647	1.08853
29	2.03160	2.09554	2.02728	2.08284	2.03006	1.97050	1.74214	1.40817
28	2.47503	2.51608	2.42815	2.47251	2.39849	2.35249	2.13070	1.71160
27	2.92035	2.91190	2.81366	2.83015	2.73560	2.70665	2.51943	1.99557
26	3.30573	3.26169	3.14909	3.14326	3.03292	3.10904	2.85679	2.24677
25	3.62239	3.55634	3.42766	3.40667	3.28467	3.26297	3.13550	2.45984
24	3.87483	3.79450	3.65082	3.61892	3.48862	3.49691	3.35941	2.63443
23	4.07020	3.97977	3.82177	3.78081	3.64397	3.66146	3.53382	2.77310
22	4.21560	4.11717	3.94475	3.89130	3.75080	3.77789	3.66390	2.87921
21	4.31721	4.21125	4.02341	3.95468	3.80815	3.84949	3.75407	2.95647
20	4.38117	4.26451	4.06170	3.97206	3.81322	3.87756	3.80856	3.00839
19	4.41319	4.28511	4.06400	3.94532	3.76060	3.86359	3.83159	3.03877
18	4.41949	4.27936	4.03658	3.87915	3.64612	3.81296	3.82874	3.05159
17	4.40628	4.25419	3.98812	3.78598	3.48462	3.73620	3.80735	3.05144
16	4.37980	4.21664	3.92952	3.68570	3.32101	3.65236	3.77635	3.04286
15	4.34518	4.17292	3.87040	3.59728	3.20169	3.57921	3.74347	3.02991
14	4.30594	4.12698	3.81643	3.52750	3.12317	3.52247	3.71324	3.01492
13	4.26326	4.08048	3.76922	3.47755	3.07526	3.48332	3.68660	2.99855
12	4.21587	4.03362	3.72751	3.44461	3.05301	3.45823	3.66210	2.93362
11	4.15991	3.98230	3.68778	3.42511	3.05704	3.44371	3.63598	2.95516
10	4.08950	3.91932	3.68778	3.42511	3.05704	3.44371	3.63598	2.95516
9	3.99627	3.83585	3.58338	3.39531	3.14120	3.41612	3.54843	2.86770
8	3.86985	3.72095	3.49362	3.34927	3.16520	3.36837	3.46317	2.78900
7	3.69891	3.56247	3.35910	3.25113	3.11259	3.26743	3.33127	2.67425
6	3.47127	3.34822	3.16705	3.8640	2.97401	3.09895	3.13965	2.51346
5	3.17803	3.06885	2.90816	2.84631	2.75063	2.85416	2.87796	2.29657
4	2.81020	2.71226	2.57666	2.52435	2.44492	2.52665	2.53976	2.01328
3	2.36765	2.26652	2.17338	2.11143	2.06225	2.10872	2.12757	1.65263
2	1.91001	1.70278	1.75459	1.58358	1.65747	1.57830	1.70207	1.20225
1	1.05685	.92831	.97056	.86197	.91469	.85773	.93536	.64098

Table 3-D Continued

EXPOSURE BY BANK

(I = 3)

K/J	1	2	3	4	5	6	7	8
32	.40270	.65181	.63021	.64878	.61850	.59902	.36749	.26458
31	.83939	1.22385	1.19310	1.22008	1.16909	1.12776	.77404	.52770
30	1.34077	1.61559	1.66342	1.61913	1.62646	1.49387	1.25315	.78842
29	1.75953	2.02728	2.09138	2.02967	2.03945	1.87917	1.67029	1.03610
28	2.19227	2.42815	2.48977	2.42384	2.41939	2.35212	2.10035	1.27617
27	2.76420	2.81366	2.85753	2.79917	2.76602	2.60773	2.64027	1.51062
26	3.17021	3.14909	3.17953	3.12346	3.06834	2.91767	3.03887	1.71268
25	3.47855	3.42766	3.44936	3.39102	3.32093	3.17572	3.34721	1.87874
24	3.71609	3.65082	3.66597	3.60300	3.52205	3.38254	3.58805	2.01152
23	3.89660	3.82177	3.83079	3.76101	3.67168	3.53990	3.77335	2.11460
22	4.02753	3.94475	3.94348	3.86660	3.76847	3.64939	3.90998	2.19155
21	4.11251	4.02341	4.00759	3.91909	3.80736	3.71085	4.00257	2.24536
20	4.15722	4.05172	4.02507	3.91548	3.77347	3.72203	4.05439	2.24536
19	4.16838	4.06400	3.99796	3.85013	3.62893	3.67173	4.06893	2.29463
18	4.15220	4.03658	3.93104	3.71922	3.27693	3.57359	4.05117	2.29609
17	4.11641	3.98812	3.83682	3.53991	2.70464	3.42374	4.01072	2.28711
16	4.06922	3.92952	3.73531	3.5739	2.13919	3.27398	3.96124	2.27226
15	4.01810	3.87040	3.64562	3.22328	1.90823	3.16455	3.91563	2.25545
14	3.96770	3.81643	3.57453	3.13468	1.81508	3.09434	3.87906	2.23904
13	3.92028	3.76922	3.52324	3.08168	1.77618	3.05404	3.85221	2.22358
12	3.87616	3.72751	3.48934	3.05933	1.77412	3.03763	3.83339	2.20825
11	3.83017	3.68778	3.46900	3.06816	1.83078	3.04567	3.81865	2.19076
12	3.87616	3.72751	3.48934	3.05933	1.77412	3.03763	3.83339	2.20825
11	3.83017	3.68778	3.46900	3.06816	1.83078	3.04568	3.81865	2.19076
10	3.77544	3.64331	3.45630	3.11117	2.04852	3.08175	3.80113	2.16724
9	3.70237	3.58338	3.43772	3.17370	2.46053	3.13305	3.76846	2.13191
8	3.59893	3.49362	3.39081	3.21015	2.87790	3.15731	3.70205	2.07729
7	3.45211	3.35910	3.29146	3.16758	2.99056	3.10478	3.58135	1.99509
6	3.24979	3.16705	3.12498	3.03506	2.91852	2.96592	3.39047	2.87697
5	2.98109	2.98109	2.90816	2.88232	2.81309	2.72354	2.74117	3.11625
4	2.63622	2.57666	2.55666	2.50348	2.42712	2.43228	2.74589	1.50079
3	2.20326	2.17338	2.13865	2.11047	2.03219	2.04504	2.27006	1.22609
2	1.65455	1.75459	1.60398	1.68651	1.52204	1.63635	1.66713	.88280
1	.90163	.97056	.87310	.92886	.82731	.90013	.89499	.46730

Table 3-D Continued

EXPOSURE BY BANK

(I = 4)

K/J	1	2	3	4	5	6	7	8
32	.64333	.63294	.64878	.38738	.59067	.51492	.39782	0.0
31	1.20758	1.19553	1.22008	.80534	1.11258	.98018	.77996	0.0
30	1.59527	1.66153	1.61913	1.28432	1.47689	1.37690	1.14100	0.0
29	1.99785	2.08284	2.02967	1.68027	1.85306	1.73673	1.47391	0.0
28	2.38660	2.47251	2.42384	2.08601	2.21384	2.06764	1.78658	0.0
27	2.75884	2.83015	2.79917	2.62469	2.55740	2.36830	2.08279	0.0
26	3.08184	3.14326	3.12346	2.99822	2.85298	2.63029	2.33757	0.0
25	3.34992	3.40667	3.39102	3.27654	3.09582	2.84928	2.54789	0.0
24	3.56457	3.61892	3.60300	3.48471	3.28749	3.02457	2.71621	0.0
23	3.72837	3.78081	3.76101	3.63501	3.42970	3.15735	2.84578	0.0
22	3.84454	3.89130	3.86660	3.73213	3.52328	3.24882	2.93984	0.0
21	3.91592	3.95468	3.91909	3.77702	3.56689	3.29999	3.00102	0.0
20	3.94557	3.97206	3.91548	3.76829	3.55691	3.31121	3.03195	0.0
19	3.93721	3.94532	3.85013	3.70478	3.48720	3.28357	3.03568	0.0
18	3.89713	3.87915	3.71922	3.58976	3.35356	3.22092	3.01695	0.0
17	3.83518	3.78598	3.53911	3.44042	3.17236	3.13472	2.98276	0.0
16	3.37643	3.68570	3.35739	2.28832	2.99140	3.04410	2.94208	0.0
15	3.69631	3.59728	3.22328	3.16414	2.85978	2.96800	2.90298	0.0
14	3.63763	3.52750	3.13468	3.07716	2.77523	2.91143	2.87002	0.0
13	3.59000	3.47755	3.08168	3.02500	2.72776	2.87402	2.84454	0.0
12	3.55217	3.44461	3.05933	3.00417	2.71223	2.85368	2.82552	0.0
11	3.52050	3.42511	3.06816	3.01226	2.72922	2.84764	2.80972	0.0
10	3.48788	3.41316	3.11117	3.04406	2.78193	2.85054	2.79127	0.0
9	3.44250	3.39531	3.17370	3.08227	2.85593	2.84979	2.76085	0.0
8	3.36816	3.34927	3.21015	3.09436	2.90586	2.82419	2.70595	0.0
7	3.24809	3.25113	3.16758	3.04558	2.87916	2.75102	2.61314	0.0
6	3.06886	3.08640	3.03506	2.91902	2.76557	2.61680	2.47033	0.0
5	2.82147	2.84631	2.81309	2.70788	2.56633	2.41345	2.26711	0.0
4	2.50074	2.52435	2.50348	2.40860	2.28385	2.13472	1.99253	0.0
3	2.10726	2.11143	2.11047	2.01488	1.92308	1.77286	1.63507	0.0
2	1.69366	1.58358	1.68651	1.50719	1.53409	1.31280	1.18268	0.0
1	.93503	.86197	.92886	.81866	.84337	.70816	.62833	0.0

Table 3-D Continued

EXPOSURE BY BANK

(I = 5)

K/J	1	2	3	4	5	6	7	8
32	.58964	.67861	.61850	.59067	.53124	.50436	.28625	0.0
31	1.11209	1.26007	1.16909	1.11258	1.00344	.93828	.54934	0.0
30	1.54221	1.63841	1.62646	1.47689	1.39363	1.21802	.77944	0.0
29	1.93148	2.03006	2.03945	1.85306	1.74302	1.50772	.98674	0.0
28	2.29153	2.39849	2.41939	2.21384	2.06120	1.77649	1.17468	0.0
27	2.62177	2.73560	2.76602	2.55740	2.34836	2.01840	1.34338	0.0
26	2.91289	3.03292	3.06834	2.85298	2.59771	2.22935	1.48954	0.0
25	3.15944	3.28467	3.32093	3.09582	2.80576	2.40694	1.61188	0.0
24	3.35775	3.48861	3.52205	3.28749	2.97167	2.55077	1.71049	0.0
23	3.50869	3.64397	3.67168	3.42970	3.09612	2.66097	1.78644	0.0
22	3.61399	3.75080	3.76847	3.52328	3.17965	2.75842	1.84093	0.0
21	3.67493	3.80815	3.80736	3.56689	3.22264	2.78414	1.87546	0.0
20	3.69229	3.81322	3.77347	3.55691	3.22496	2.79975	1.89169	0.0
19	3.66775	3.76060	3.62893	3.48720	3.18745	2.78770	1.89189	0.0
18	3.60572	3.64612	3.27603	3.35356	3.11407	2.75271	1.87908	0.0
17	3.51820	3.48462	2.70464	3.17236	3.01693	2.70233	1.85743	0.0
16	3.42431	3.32101	2.13919	2.99140	2.91667	2.64672	1.83159	0.0
15	3.34232	3.20169	1.90823	2.85978	2.83235	2.59502	1.80586	0.0
14	3.27890	3.12317	1.81508	2.77523	2.76975	2.55258	1.78299	0.0
13	3.23377	3.07526	1.77618	2.72776	2.72905	2.52099	1.76408	0.0
12	3.20399	3.05301	1.77412	2.71223	2.70825	2.49935	1.74857	0.0
11	3.18663	3.05704	1.83072	2.72922	2.70454	2.48450	1.73439	0.0
10	3.17607	3.09082	2.04852	2.78193	2.71235	2.47059	1.71777	0.0
9	3.15944	3.14120	2.46053	2.85593	2.71865	2.44817	1.69323	0.0
8	3.11542	3.16520	2.87790	2.90586	2.70164	2.40454	1.65390	0.0
7	3.02132	3.11259	2.99056	2.87916	2.63822	2.32650	1.59234	0.0
6	2.86393	2.97401	2.91852	2.76557	2.51507	2.20336	1.50157	0.0
5	2.63595	2.75063	2.72354	2.56633	2.32476	2.02756	1.37516	0.0
4	2.33280	2.44492	2.42712	2.28385	2.06218	1.79455	1.20721	0.0
3	1.94735	2.06225	2.03219	1.92308	1.72079	1.50549	.99188	0.0
2	1.45913	1.65747	1.52204	1.53409	1.28469	1.20189	.72226	0.0
1	.79324	.91469	.82731	.84337	.69663	.64945	.38485	0.0

EXPOSURE BY BANK

(I = 6)

K/J	1	2	3	4	5	6	7	8
32	.62364	.58975	.59902	.51492	.50436	.33508	0.0	0.0
31	1.17440	1.11833	1.12776	.98018	.93828	.68285	0.0	0.0
30	1.55925	1.56268	1.49387	1.37690	1.21802	.87788	0.0	0.0
29	1.96346	1.97050	1.87917	1.73673	1.50772	1.09317	0.0	0.0
28	2.35724	2.35249	2.25212	2.06764	1.77649	1.28498	0.0	0.0
27	2.73554	2.70665	2.60773	2.36830	2.01840	1.45432	0.0	0.0
26	3.06697	3.01904	2.91767	2.63029	2.22935	1.60063	0.0	0.0
25	3.34466	3.28297	3.17572	2.84928	2.40694	1.72356	0.0	0.0
24	3.56944	3.49691	3.38254	3.02457	2.55077	1.82302	0.0	0.0
23	3.74377	3.66146	3.53990	3.15735	2.66097	1.89954	0.0	0.0
22	3.87109	3.77789	3.64939	3.24882	2.73842	1.95383	0.0	0.0
21	3.95441	3.84949	3.71085	3.29999	2.78414	1.98699	0.0	0.0
20	3.99691	3.87756	3.72203	3.31121	2.79975	2.00043	0.0	0.0
19	4.00231	3.86395	3.67773	3.28357	2.78770	1.99644	0.0	0.0
18	3.97670	3.81296	3.57359	3.22092	2.75271	1.97829	0.0	0.0
17	3.92954	3.73620	3.42374	3.13472	2.70233	1.95064	0.0	0.0
16	3.87330	3.65236	3.27198	3.04410	2.64672	1.91878	0.0	0.0
15	3.81905	3.57921	3.16355	2.96800	2.59502	1.88762	0.0	0.0
14	3.77292	3.52247	3.09434	2.91143	2.55258	1.86035	0.0	0.0
13	3.73626	3.48332	3.05404	2.87402	2.52099	1.83831	0.0	0.0
12	3.70743	3.45823	3.03763	2.85368	2.49935	1.82095	0.0	0.0
11	3.68255	3.44371	3.04567	2.84764	2.48450	1.80602	0.0	0.0
10	3.65425	3.43402	3.08175	2.85054	2.47059	1.78943	0.0	0.0
9	3.61047	3.41612	3.13305	2.84979	2.44817	1.76519	0.0	0.0
8	3.53467	3.36837	3.15731	2.82419	2.40454	1.72585	0.0	0.0
7	3.40958	3.26743	3.10478	2.75102	2.32650	1.66351	0.0	0.0
6	3.22111	3.09895	2.96592	2.61680	2.20336	1.57103	0.0	0.0
5	2.95941	2.85416	2.74117	2.41345	2.02756	1.44222	0.0	0.0
4	2.61882	2.52665	2.43228	2.13472	1.79455	1.27184	0.0	0.0
3	2.20113	2.10872	2.04504	1.77286	1.50549	1.05441	0.0	0.0
2	1.76339	1.57830	1.63635	1.31280	1.20189	.78019	0.0	0.0
1	.97156	.85773	.90013	.70816	.65945	.42026	0.0	0.0

EXPOSURE BY BANK

(I = 7)

K/J	1	2	3	4	5	6	7	8
32	.41602	.52115	.36749	.39782	.28625	0.0	0.0	0.0
31	.87286	.99945	.77404	.77996	.54934	0.0	0.0	0.0
30	1.40648	1.35641	1.25315	1.14100	.77944	0.0	0.0	0.0
29	1.87265	1.74214	1.67029	1.47391	.98674	0.0	0.0	0.0
28	2.35648	2.13070	2.10035	1.78658	1.17468	0.0	0.0	0.0
27	2.96341	2.51943	2.64027	2.08279	1.34338	0.0	0.0	0.0
26	3.41646	2.85679	3.03887	2.33757	1.48954	0.0	0.0	0.0
25	3.77057	3.13550	3.34721	2.54789	1.61188	0.0	0.0	0.0
24	4.05047	3.35941	3.58805	2.71621	1.71049	0.0	0.0	0.0
23	4.26982	3.53382	3.77335	2.84578	1.78644	0.0	0.0	0.0
22	4.43705	3.66390	3.90998	2.93984	1.84093	0.0	0.0	0.0
21	4.55871	3.75407	4.00257	3.00102	1.87546	0.0	0.0	0.0
20	4.63985	3.80856	4.05439	3.03195	1.89169	0.0	0.0	0.0
19	4.68614	3.83159	4.06893	3.03568	1.89189	0.0	0.0	0.0
18	4.70360	3.82874	4.05117	3.01695	1.87908	0.0	0.0	0.0
17	4.70024	3.80735	4.01072	2.98276	1.85743	0.0	0.0	0.0
16	4.68421	3.77635	3.96124	2.94208	1.83159	0.0	0.0	0.0
15	4.66287	3.74347	3.91563	2.90298	1.80586	0.0	0.0	0.0
14	4.64072	3.71324	3.87906	2.87002	1.78299	0.0	0.0	0.0
13	4.61928	3.68660	3.85221	2.84454	1.76408	0.0	0.0	0.0
12	4.59684	3.66210	3.83339	2.82552	1.74857	0.0	0.0	0.0
11	4.56905	3.63598	3.81865	2.80972	1.73439	0.0	0.0	0.0
10	4.52790	3.60151	3.80113	2.79127	1.71777	0.0	0.0	0.0
9	4.46136	3.54843	3.76846	2.76085	1.69323	0.0	0.0	0.0
8	4.35477	3.46317	3.70205	2.70595	1.65390	0.0	0.0	0.0
7	4.19169	3.33127	3.58135	2.61314	1.59234	0.0	0.0	0.0
6	3.95586	3.13965	3.39047	2.47033	1.50157	0.0	0.0	0.0
5	3.63153	2.87796	3.11625	2.26711	1.37516	0.0	0.0	0.0
4	3.20392	2.53976	2.74689	1.99253	1.20721	0.0	0.0	0.0
3	2.65711	2.12757	2.27006	1.63507	.99188	0.0	0.0	0.0
2	1.96596	1.70207	1.66713	1.18268	.72226	0.0	0.0	0.0
1	1.06147	.93536	.89499	.62833	.38485	0.0	0.0	0.0

Table 3-D Continued

EXPOSURE BY BANK

(I = 8)

K/J	1	2	3	4	5	6	7	8
32	.38577	.38503	.26458	0.0	0.0	0.0	0.0	0.0
31	.76366	.75048	.52770	0.0	0.0	0.0	0.0	0.0
30	1.13137	1.08853	.78842	0.0	0.0	0.0	0.0	0.0
29	1.47902	1.40817	1.03610	0.0	0.0	0.0	0.0	0.0
28	1.81310	1.71160	1.27617	0.0	0.0	0.0	0.0	0.0
27	2.13567	1.99557	1.51062	0.0	0.0	0.0	0.0	0.0
26	2.41698	2.24677	1.71268	0.0	0.0	0.0	0.0	0.0
25	2.65211	2.45984	1.87874	0.0	0.0	0.0	0.0	0.0
24	2.84354	2.63443	2.01152	0.0	0.0	0.0	0.0	0.0
23	2.99546	2.77310	2.11460	0.0	0.0	0.0	0.0	0.0
22	3.11250	2.87921	2.19155	0.0	0.0	0.0	0.0	0.0
21	3.19887	2.95647	2.24536	0.0	0.0	0.0	0.0	0.0
20	3.25872	3.00839	2.27879	0.0	0.0	0.0	0.0	0.0
19	3.29601	3.03877	2.29463	0.0	0.0	0.0	0.0	0.0
18	3.31511	3.05159	2.29609	0.0	0.0	0.0	0.0	0.0
17	3.32043	3.05144	2.28711	0.0	0.0	0.0	0.0	0.0
16	3.31647	3.04286	2.27226	0.0	0.0	0.0	0.0	0.0
15	3.30692	3.02991	2.25545	0.0	0.0	0.0	0.0	0.0
14	3.29429	3.01492	2.23904	0.0	0.0	0.0	0.0	0.0
13	3.27914	2.99855	2.22358	0.0	0.0	0.0	0.0	0.0
12	3.26016	2.97962	2.20825	0.0	0.0	0.0	0.0	0.0
11	3.23410	2.95516	2.19076	0.0	0.0	0.0	0.0	0.0
10	3.19574	2.92018	2.16724	0.0	0.0	0.0	0.0	0.0
9	3.13777	2.86770	2.13191	0.0	0.0	0.0	0.0	0.0
8	3.05115	2.78900	2.07729	0.0	0.0	0.0	0.0	0.0
7	2.92557	2.67425	1.99509	0.0	0.0	0.0	0.0	0.0
6	2.74990	2.51346	1.87697	0.0	0.0	0.0	0.0	0.0
5	2.51269	2.29657	1.71496	0.0	0.0	0.0	0.0	0.0
4	2.20175	2.01328	1.50079	0.0	0.0	0.0	0.0	0.0
3	1.80324	1.65263	1.22609	0.0	0.0	0.0	0.0	0.0
2	1.30268	1.20225	.88280	0.0	0.0	0.0	0.0	0.0
1	.69176	.64098	.46730	0.0	0.0	0.0	0.0	0.0

Table 3-D Continued