

FEB 16 1989

Docket No. 50-320

GPU Nuclear Corporation

ATTN: Mr. M. B. Roche

Vice President/Director, TMI-2

P. O. Box 480

Middletown, Pennsylvania 17057

Gentlemen:

Subject: Inspection 50-320/88-20

This refers to the routine safety inspection conducted by site inspectors from December 18, 1988, to January 28, 1989 of activities authorized by NRC License No. DPR-73 and to discussions of our findings held by Messrs. D. Johnson and A. Sidpara of our office with you and other members of your staff on January 31, 1989. Based on the results of this inspection, the inspectors identified no conditions adverse to nuclear safety or regulatory requirements.

No response to this report is required unless there are misunderstandings in the accuracy of the documentation. Your cooperation with us is appreciated.

Sincerely,

**ORIGINAL SIGNED BY
LEE H. BETTENHAUSEN**

Lee H. Bettenhausen, Chief
Project Branch No. 1
Division of Reactor Projects

Enclosures:

1. NRC Region I Inspection Report 50-320/88-20
2. GPUNC - TMI-2 Clean-Up Program Status Update

cc w/encls:

T. F. Demmitt, Deputy Director, TMI-2
R. E. Rogan, Licensing and Nuclear Safety Director
J. J. Byrne, Manager, TMI-2 Licensing
S. Levin, Defueling Director
J. B. Lieberman, Esquire
W. J. Marshall, Manager, Plant Operations
Ernest L. Blake, Jr., Esquire
G. A. Kuehn, TMI-2 Site Operations Director
TMI Alert (TMIA)
Susquehanna Valley Alliance (SVA)
Public Document Room (PDR)
Local Public Document Room (LPDR)
Nuclear Safety Information Center (NSIC)
NRC Resident Inspector
Commonwealth of Pennsylvania

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PDR ADDCK 05000320
Q PDC

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IEU1

FEB 16 1989

bcc w/encs:

Region I Docket Room (with concurrences)

S. Lewis, OGC

Management Assistant, DRMA, RI (w/o encl)

DRP Section Chief, RI

Michael Masnik, NRR/PD14

Lee Thonus, NRR/PD14

Robert J. Bores, DRSS, RI

RI:DRP

DL
DJohnson
2/10/89

RI:DRP

CC
CCowgill
2/13/89

RI:DRP

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LBettenhausen
2/16/89

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11/29/80

U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-320/88-20
Docket No. 50-320
License No. DPR-73 Priority -- Category C
Licensee: GPU Nuclear Corporation
P. O. Box 480
Middletown, Pennsylvania 17057

Facility Name: Three Mile Island Nuclear Station, Unit 2

Inspection At: Middletown, Pennsylvania

Inspection Conducted: December 18, 1989 - January 28 and 31, 1989

Inspectors: R. Conte, Senior Resident Inspector
D. Johnson, Acting Senior Resident Inspector
T. Moslak, Resident Inspector
A. Sidpara, Resident Inspector (Reporting Inspector)

Approved by: C. Cowgill, Chief, Reactor Projects Section 1A

2/15/89
Date

Inspection Summary:

Areas Inspected: Routine safety inspection by site inspectors of defueling and decontamination activities, including the proper implementation of radiological controls and housekeeping measures, and licensee actions on previous inspection findings.

Results: The licensee personnel conducted the defueling activities in a safe manner. No major problems occurred. Ten previous inspection findings were closed based on inspector review for current applicability and licensee actions to resolve the issues.

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DETAILS

1.0 Overview

1.1 Licensing Activities

Following removal of the grid forging from the reactor vessel, core debris was removed from the next lower plate of the Lower Core Support Assembly (LCSA), the Incore Guide Support Plate (IGSP). Upon clearing the debris from the support plate, the two-inch thick IGSP was cut into four quadrants and completely removed from the vessel on January 12, 1989. With the IGSP out of the vessel, defueling crews are air lifting core debris found on the fifth and final plate, the elliptical flow distributor. Once this debris is removed, the flow distributor will be cut using the plasma arc technique. Removal of the flow distributor will provide access for defueling the approximately thirty tons of debris in the lower head of the reactor vessel.

1.2 NRC Staff Activities

The purpose of this inspection was to assess licensee activities during defueling and decontamination activities. The inspectors made this assessment through observations of licensee activities, interviews with licensee personnel, and review of applicable documents. NRC staff inspections use the acceptance criteria and guidance of NRC Inspection Procedures (NIP's). These NIP's were annotated in the Table of Contents to this report.

The inspectors reviewed licensee's procedures implementing control on several interfacing systems with the reactor vessel to assure adequate controls were in place to prevent uncontrolled boron dilution. The inspectors also reviewed the instrument calibration switching and tagging of valves, approval authority, and responsibilities of the operations personnel.

1.3 Persons Contacted

During this inspection, the following key licensee personnel provided substantial information in the development of the inspectors' findings.

- J. Byrne, Manager, TMI-2 Licensing
- W. County, Quality Assurance (QA) Auditor
- *J. Fornicola, Manager, TMI QA
- G. Kuehn, Site Operations Director, TMI-2
- *S. Levin, Director, Defueling
- *W. Marshall, Manager, Plant Operations
- H. Mumford, Post-Defueling Monitored Storage Manager
- M. Roche, Director, TMI-2
- *R. Rogan, Director, Licensing & Nuclear Safety, TMI-2

- *E. Schrull, TMI-2 Licensing Engineer
- *D. Turner, Director, Radiological Controls
- R. Wells, Licensing Engineer

*Attended the final management meeting.

2.0 Defueling/Decontamination Activities

2.1 Scope of Review

The inspector observed and/or reviewed licensee defueling/decontamination activities to: (1) ascertain factual status of such activities and (2) assure proper adherence to applicable procedures. The inspector also made observations in facility spaces with respect to proper housekeeping, fire protection, and radiological controls. The general acceptance criteria for this review was Section 6 of the TMI-2 Technical Specifications (TS).

In performing the above inspections, the inspectors focused on the following areas of licensee performance:

- control of operations in progress by supervisory personnel;
- knowledge of the task by technicians and support persons;
- appropriateness of governing documents, including procedures and Radiation Work Permits (RWP's);
- alertness of various controlling station personnel;
- assess the quality of implementation of selected evolutions witnessed; and,
- assess the material condition of the plant.

2.2 Emergency Medical Drill

On December 15, 1988, the annual emergency medical drill was held. An NRC regional emergency preparedness specialist observed the drill. The drill scenario involved an individual performing oxy-acetylene cutting in the turbine building in a temporary Radiation Work Permit (RWP) area. According to the drill, the torch malfunctioned and exploded in the worker's face. The worker hypothetically suffered a lacerated neck and facial burns and was supposed to be contaminated. The severity of the hypothetical injury required off-site medical assistance and transport to an off-site medical center.

Areas where the licensee performed satisfactorily included:

- medical technicians treated the patient quickly;

- an unusual event was declared and notifications were completed in a timely manner;
- the off-site ambulance crew arrived very quickly and were at the scene within minutes of arriving on site;
- security control was evident at both the accident scene and the hospital;
- contamination control techniques were adequate; and,
- posting of radiation area was evident at the hospital.

Areas where improvements can be made included:

- interface between radiation control technicians and medical technicians;
- timely identification of injured person; and,
- improved preparation for the drill; the "injured" person did not have any protective clothing (PC) and, thus, the removal of the contaminated PC's was not performed.

In summary, licensee performance was adequate to protect the health and safety of the public. Licensee action on previously-identified weaknesses was also adequate.

2.3 Other General Findings

As a result of the routine review noted above, the inspectors identified no major discrepancies. In general, licensee representatives properly implemented procedures. Defueling activities and clean-up activities were conducted in a safe and controlled manner. Repair activities required for the plasma arc torches and reactor vessel cutting equipment were accomplished appropriately. There were no specific events or incidents that the NRC staff reviewed during this inspection period.

No unacceptable conditions were identified.

3.0 TMI-2 Clean-Up Program Status Meeting

3.1 Overview

The NRC staff met with the licensee staff in the Region I office on January 11, 1989, to discuss the status of on-going defueling and decontamination activities conducted at Unit 2. Senior site management presented information on the topics of schedule, budget, defueling status, defueling completion reports, water disposal status, Special Nuclear

Material Measurement (SNMM) program, radiological conditions, and proposed criteria for eventually entering Post-Defueling Monitored Status (PDMS).

Information sheets provided by the licensee are in Attachment 1 to this report. NRC and licensee staff attending this meeting are listed in Attachment 2.

3.2 Decontamination Efforts

The licensee is concentrating its resources to complete defueling of the reactor vessel and primary system by the summer of 1989. To achieve this goal, efforts to decontaminate the auxiliary and fuel handling buildings are being deferred until defueling is accomplished. This shift in emphasis from performing defueling and decontamination efforts in parallel to completing these activities one after the other is intended to minimize the need to repeat decontamination and to better utilize the available resources.

3.3 Exposure Status

The licensee has met or exceeded short-term and collective goals for controlling radiation exposure to personnel throughout the decontamination and defueling effort. Total worker exposure from March 28, 1979 through December 31, 1988, was 5,541 person-rem. It is currently estimated that a cumulative total of approximately 6,000 person-rem will be expended for the entire clean-up. During 1988, total person-rem expended was approximately 12 percent of the estimated exposure of 994 person-rem.

The licensee attributed these exposure results to proper ALARA (as low as reasonably achievable) planning. The NRC staff considered that the programmatic exposure control efforts to be effective.

3.4 Data Management

The NRC staff was briefed on the licensee's program to measure and document the residual fuel that may remain in plant systems following completion of clean-up activities. The various measurement techniques used to quantify the SNM were discussed. Post-Defueling Survey Records and the Defueling Completion Reports will be the measurement documentation that provide the basis for the licensee's final SNM (fuel) accountability.

3.5 Meeting Summary

The licensee provided detailed information on the progress of clean-up activities. It was agreed that future meetings of this nature would be useful for assessing the overall status of TMI-2.

4.0 Licensee Action on Previous Inspection Findings

4.1 (Closed) Unresolved Item (320/83-07-02): Air Intake Tunnel (AIT) Halon System Intentional Deactivation

The licensee, on several occasions, intentionally deactivated the AIT to prevent inadvertent actuation by lightning. The objective behind the deactivation was based on the assessment that reactivation of the AIT following a thunderstorm would take significantly less time than recharging system following a spurious discharge. This mode of the operation placed the plant in the Action Statement of the Technical Specification (TS) Section 3.7.10.3, which required restoration of the system to operable status within fourteen days. The licensee also prepared the required Licensee Event Reports (LER's). The inspector, following review of the LER's, determined that the corrective actions requiring immediate restoration of the AIT system to operable status and installation of louvers to protect the AIT system detectors from exposure to the lightning were adequate. This item is closed.

4.2 (Closed) Unresolved Item (320/83-08-06): Work Under Expired Special Operating Procedures

This item involved operation of the two containment isolation valves DSA-V-004 and DSA-V-005 under expired Special Operating Procedure (SOP). Operation of these valves without a valid procedure violated TS Section 3.6.1 requirements and was reportable pursuant to TS 6.9.1.8(b). The licensee issued a LER, which was later updated. The licensee additionally identified a total of sixteen such violations and determined the root cause to be the failure of operations personnel to include these valves in the SOP, as well as failure to follow the administrative procedure governing validation of SOP's. The NRC review of the LER indicated the immediate corrective action of discontinuing operation of these valves until a new SOP was issued and the long-term corrective action to counsel operations personnel emphasizing performance of activities in accordance with approved procedures were adequate. No further actions were required. This item is closed.

4.3 (Closed) Unresolved Item (320/83-12-01): Method for Control of Locked Valve List

The inspector identified some weaknesses in the area of control of locked valves, including authority to initiate changes and assignment of valves to the systems as well as maintenance of log sheets. In response, the licensee revised Administrative Procedure (AP) 4210-ADM-3020.05, "Controlled Key Locker." This procedure addressed inventory of keys for important-to-safety facilities; system interlocks and locked valves; and, responsibilities, authorities, and maintenance of required logs. The inspector, following review of this procedure, determined that the procedure was adequate and there was no need for further actions. This item is closed.

4.4 (Closed) Inspector Follow Item (320/83-12-03): Control of Safety and Non-Safety-Related Welding

The inspector identified a concern that the licensee's Administrative Procedure (AP) 1063, Revision 1, "Control of Welding," did not require site welding engineering personnel to review welding qualification documentation for safety and non-safety-related equipment. This problem was corrected by revising the procedure incorporating the requirement. The inspector also had concerns about the interpretations of Weld Procedure Specification (WPS) and Procedure Qualification Report (PQR) documentations, specifically regarding the qualified thickness ranges for American Society of Mechanical Engineers (ASME) and American Welding Society (AWS) welding specifications. The licensee stated that the welding program required all WPS's to be qualified in accordance with Section IX of the ASME Code. NRC review of licensee documents indicated satisfactory response to the concerns. This item is closed.

4.5 (Closed) Deviation (320/83-14-01): Independent Design Verification by Supervisors

This issue dealt with an administrative weakness in the area of supervisors performing independent design verification where the design was performed by subordinate engineers. In response, the licensee revised procedure 4000-ENG-7310.3, "Engineered Document Verification Procedure." The inspector determined that the licensee's revised procedure was adequate. This item is closed.

4.6 (Closed) Inspector Follow-Up (320/83-14-02): Informal Transfer of Work Between Plant and Site Engineering

This item involved an administrative weakness concerning informal transfer of work between plant engineering and site engineering even though both groups had good communication and the informality did not result in any problem. The licensee, in order to strengthen the current engineering controls, revised AP 4000-ADM-7350.02, "Engineering Service Request," to add formal instructions. Following review of this procedure, the inspector determined the corrective actions to be adequate. This item is closed.

4.7 (Closed) Violation (320/83-19-01): Inadequate Precautions to Prevent Overpressurization of Piping System

The violation occurred during flushing of the discharge line between the concentrated waste storage tank (CWST) and the associated liquid waste pump. The flushing pressure of 56 psig exceeded the bursting pressure limit of the rupture disc in the line. The root cause was determined to be the failure to provide appropriate precautions or limits in the Unit Work Instruction (UWI) 4220-3233-83-0768. The licensee's corrective actions included review of the incident with the appropriate personnel, as well as combining of two Maintenance Procedures (MP's) 1410-Y-17,

"Cleaning of Clogged Lines," and 1430-Y-24, "Clearing Instrument Lines of Moisture, Blockage, or Lowering Radioactive Levels," into a single new procedure 4220-CMG-3921.03, "Clearing All Lines of Moisture, Blockage, or Lowering Radioactivity." The inspector, following review of this procedure determined that it included necessary precautions and instruction to establish maximum allowable pressure for flushing. The licensee response was adequate. This violation is closed.

4.8 (Closed) Bulletin (83-BU-03): Check Valve Failures in Raw Water Cooling of Emergency Diesel Generators

This bulletin dealt with the failures of check valves in the raw water cooling system of diesel generators at the Dresden and Quad-Cities nuclear plants, as well as other related events. In response to this bulletin, the licensee reviewed the existing testing and surveillance program for similar valves and addressed all the concerns identified in the bulletin. Additionally, the licensee tested the affected valves to verify operability and did not identify any problem. The licensee response was timely and adequate. The inspector also noted that the two emergency diesel generators at Unit-2 were not required by the current TS. This bulletin's requirements were satisfied and it was closed.

4.9 (Closed) Bulletin (83-BU-07): Fraudulent Products Sold by Ray Miller, Inc.

The bulletin provided a comprehensive list of customers who had apparently received fraudulent products from Ray Miller, Inc. for the years 1975 through 1979. The licensee reviewed the list and identified one customer who supplied some materials that were originally received from Ray Miller, Inc. However, the licensee determined that the material was not part of any safety-related system and it did not have any safety significance. The licensee also provided information on the procured items directly from Ray Miller, Inc. beyond the period specified in the bulletin. The licensee's assessment on each such item did not present any problem. The inspector reviewed the licensee response and determined that it was adequate and no further actions were required. This bulletin was closed.

4.10 (Closed) Unresolved Item (320/84-09-02): Purging of Containment with One Isolation Valve Inoperable

Based upon the review of operations logs, the NRC inspector discovered that the licensee purged the containment using train "A" even though the isolation valve AH-V-4A was inoperable. This action did not comply with TS Section 3.6.1.1, which required that for each containment penetration, two isolation valves be maintained operable and closed unless allowed open pursuant to an approved procedure. It also required that with one valve open or inoperable the second isolation valve should be deactivated and secured in the isolation position. In this event, valve AH-V-4A (outside of containment) failed to close following a surveillance test; however, the purging of containment continued. This event was reportable pursuant to 10 CFR 50.73(a)(2)(i)(B) and, therefore, the licensee issued

Licensee Event Report (LER) 84-07. Review of the LER indicated that the event was caused by misinterpretation of TS requirements. The corrective action involved deactivation of the isolation valve AH-V-3A (inside containment) as required by the TS, as well as review of the incident with operations personnel. The inspector did not have any further concern and determined that the licensee actions were adequate.

5.0 Management Meeting

The inspector discussed the inspection scope and findings with licensee management periodically during the course of the inspection and at a final meeting conducted January 31, 1989. Licensee management personnel attending the final exit meeting is noted in paragraph 1.3.

The inspection results, as discussed at the meeting, are summarized in the cover page of the inspection report. Licensee representatives indicated that none of the subjects discussed contained proprietary or safeguards information.

Unresolved Items are matters about which information is required in order to ascertain whether they are acceptable, violations, or deviations. These items are addressed in Section 4.0.

Inspector Follow Items are matters which were established to administratively follow open issued based on inspector judgement or on licensee/staff commitment. These are addressed in Section 4.0.

TMI - 2
CLEAN UP PROGRAM
STATUS UPDATE

JANUARY 11, 1989

8902280295 890216
PDR ADOCK 05000320
Q PDR

TMI-2
CLEANUP PROGRAM
STATUS UPDATE

NRC
REGION I

January 11, 1989

MEETING WITH NRC REGION I

AGENDA

<u>Subject</u>	<u>Presenter</u>	<u>Time</u>
Introduction	M. B. Roche	5 min.
Schedule and Budget	M. B. Roche	15 min.
Defueling Status	S. Levin	15 min.
Defueling Completion Report	J. J. Byrne	5 min.
Water Disposal Status	G. A. Kuehn	15 min.
SNM Measurement Program	J. S. Schork	15 min.
Radiological Conditions	D. W. Turner	15 min.
PDMS (Mode 4) Proposal	R. E. Rogan	30 min.

TMI-2

DIRECTOR'S COMMENTS

INTRODUCTION

PROGRAM STATUS

MILESTONES

BUDGET

January 9, 1989

SCHEDULE MILESTONES

DEFUELING

SUBMIT DEFUELING COMPLETION REPORT	JUNE 1, 1989
COMPLETE IN-VESSEL DEFUELING	JUNE 30, 1989
COMPLETE NRC LOWER HEAD SAMPLING	JULY 31, 1989
ENTER MODE 2 (DEFUELING COMPLETE)	JULY 31, 1989
COMPLETE REACTOR VESSEL & RCS DRAIN	SEPT. 15, 1989

DECONTAMINATION

COMPLETE RB DECON	DEC. 31, 1989
COMPLETE AFH3 DECON	AUG. 31, 1990

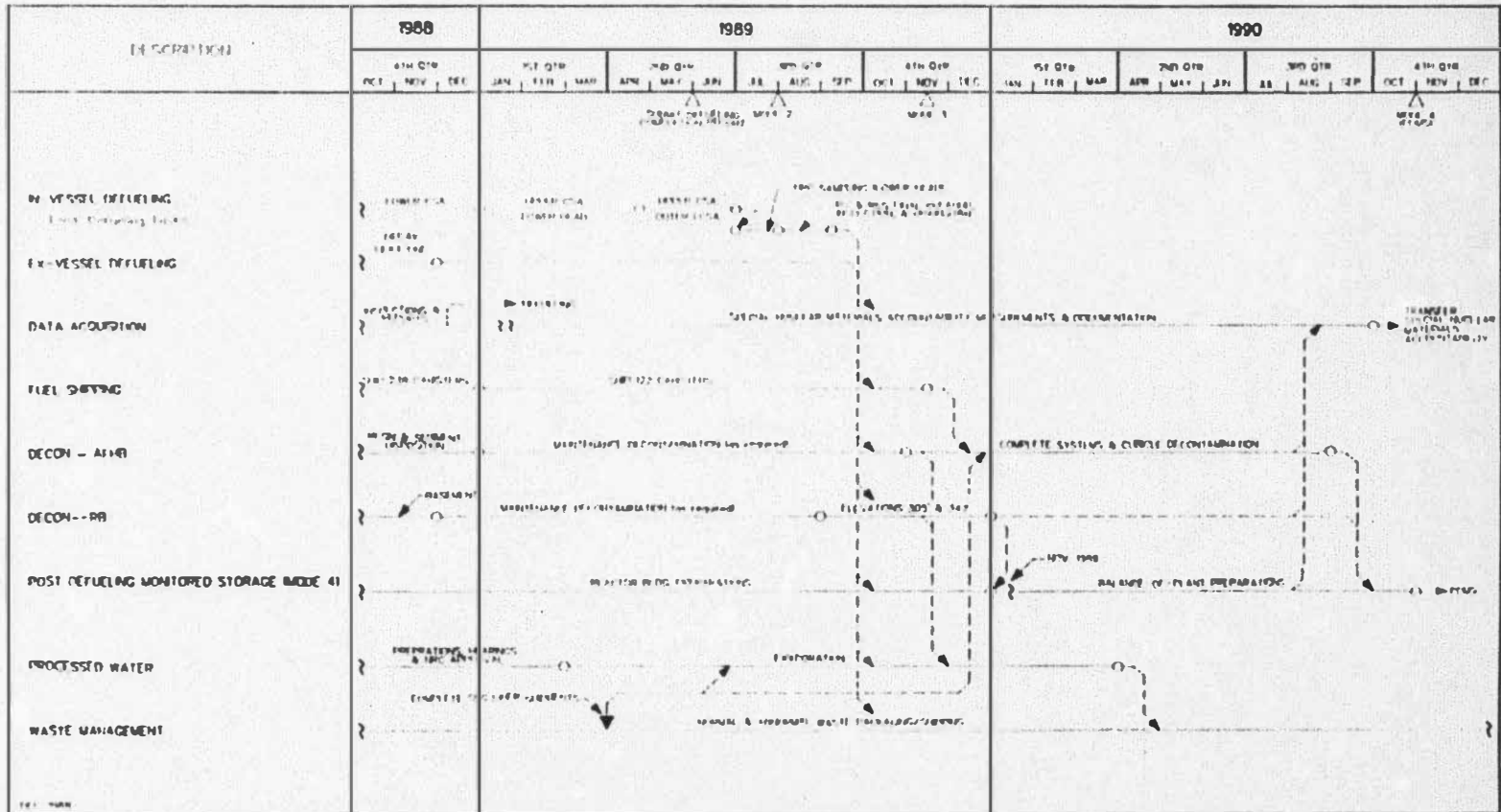
FUEL SHIPPING

COMPLETE FUEL SHIPPING	NOV. 15, 1989
ENTER MODE 3 (FUEL SHIPPING COMPLETE)	NOV. 15, 1989

OTHER ACTIVITIES

START PROCESSED WATER EVAPORATION	FEB. 28, 1989
COMPLETE PROCESSED WATER EVAPORATION	MAR. 31, 1990
COMPLETE TRANSFER OF SNM ACCOUNTABILITY	SEPT. 30, 1990
COMPLETE MONITORED STORAGE PREREQUISITES	OCT. 31, 1990

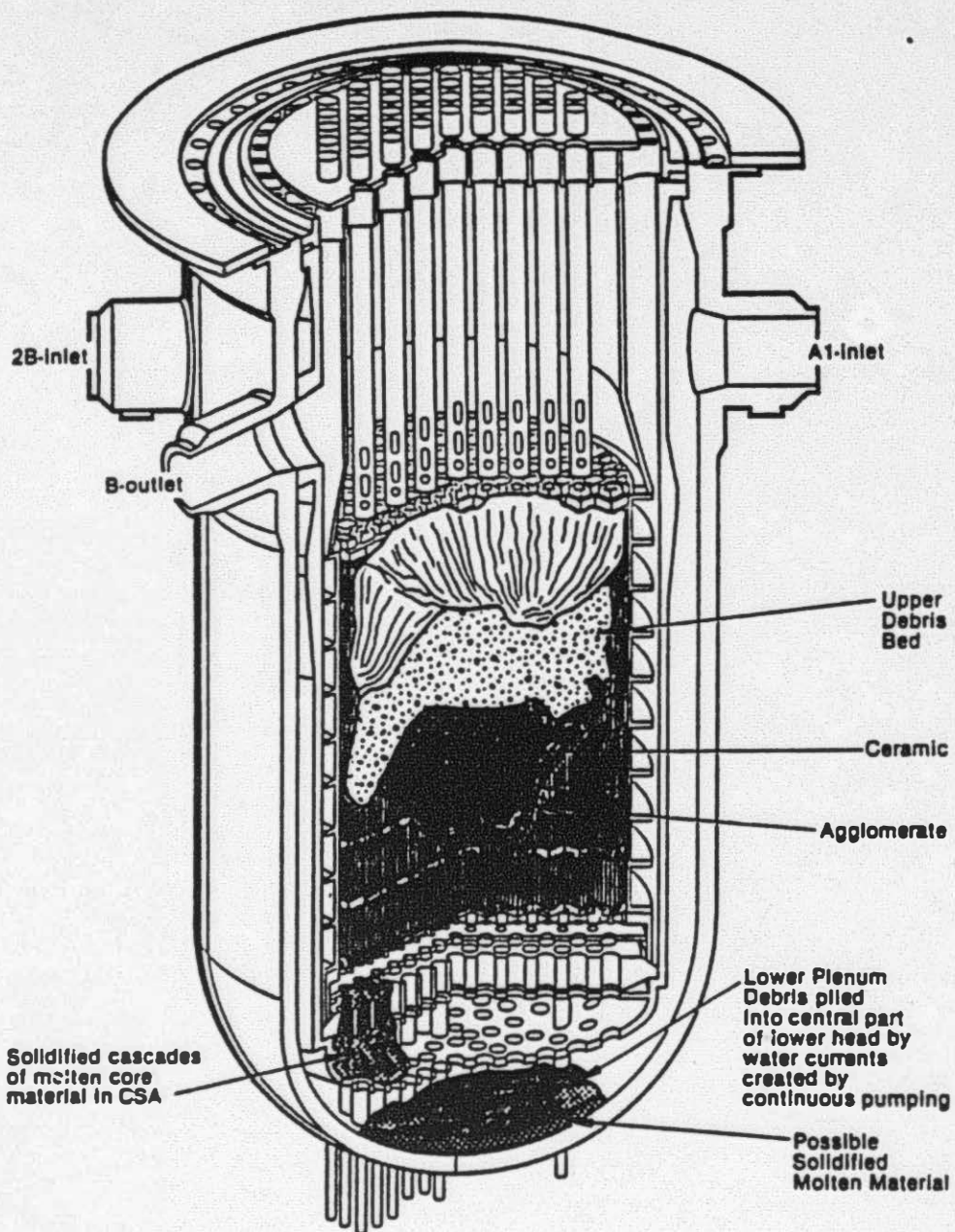
TMI-2 PROGRAM MASTER SCHEDULE



TMI-2

DEFUELING STATUS

UPDATE



LATEST EG&G DIAGRAM OF DAMAGED FUEL

REACTOR COOLANT SYSTEM COMPONENTS

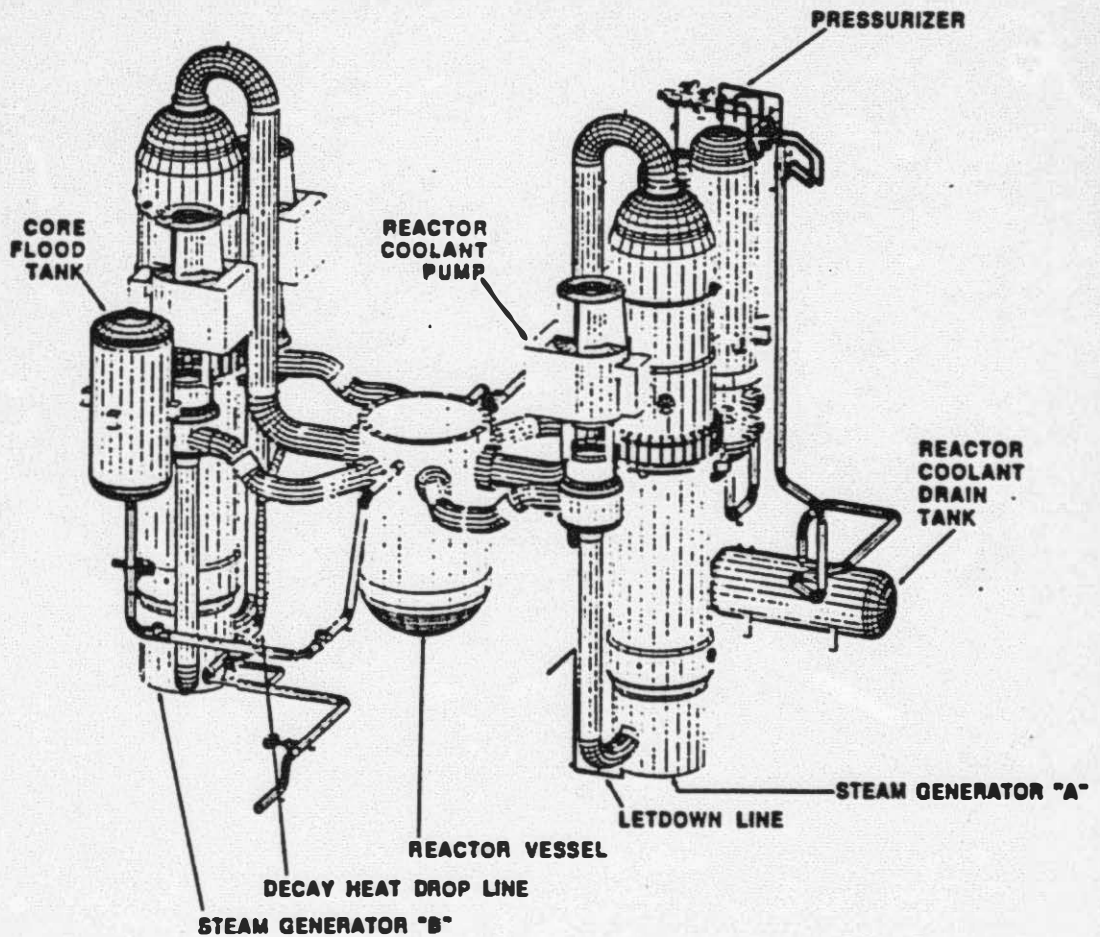


TABLE 3

CATEGORIZATION OF DEFUELING/DWCS CANISTERS AS OF 1/6/89

TYPE OF CANISTERS (Use)	DEFUELING(a)		DWCS(b)		TOTAL		SHIPPED		IN FHB POOL-A	
	NUMBER	WEIGHT (LB)	NUMBER	WEIGHT (LB)	NUMBER	WEIGHT (LB)	NUMBER	WEIGHT (LB)	NUMBER	WEIGHT (LB)
FUEL	215	206,352	0	0	215	206,352	202	195,831	13	10,521
KNOCKOUT	2	1,378	5	25	7	1,403	6	1,123	1	280
FILTER	1	37	30	322	31	359	30	322	1	37
TOTALS:	218	207,767	35	347	253	208,114	238	197,276	15	10,838
PERCENT OF TOTAL (c)		70.0%		0.1%		70.1%		66.4%		3.7%

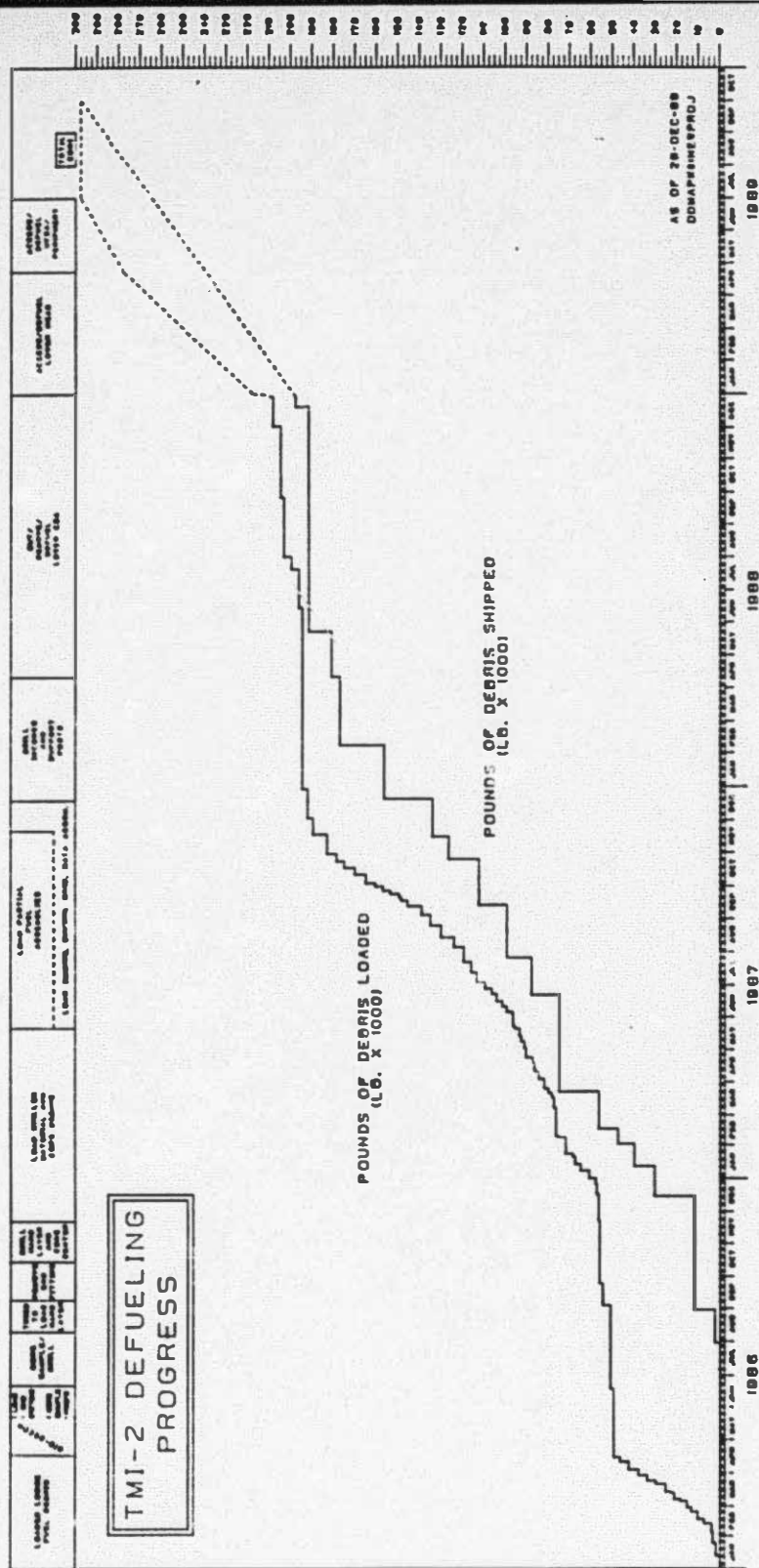
REMAINING WEIGHT TO BE TRANSFERRED(c): 88,786 (LB)

(a) INCLUDES 5 CORE BORE CANISTERS SHIPPED TO INEL.

(b) THIS INCLUDES ONLY DWCS CANISTERS SHIPPED TO INEL.

(c) BASED ON AN ESTIMATED TOTAL INVENTORY OF 296,900 LB.

DATE: 1/6/89
BY: R. RAINISCH



ESTIMATED CURRENT CORE MATERIAL DISTRIBUTION

Originally Estimated Post-Accident Core Material

293,100 lb

Additional Material

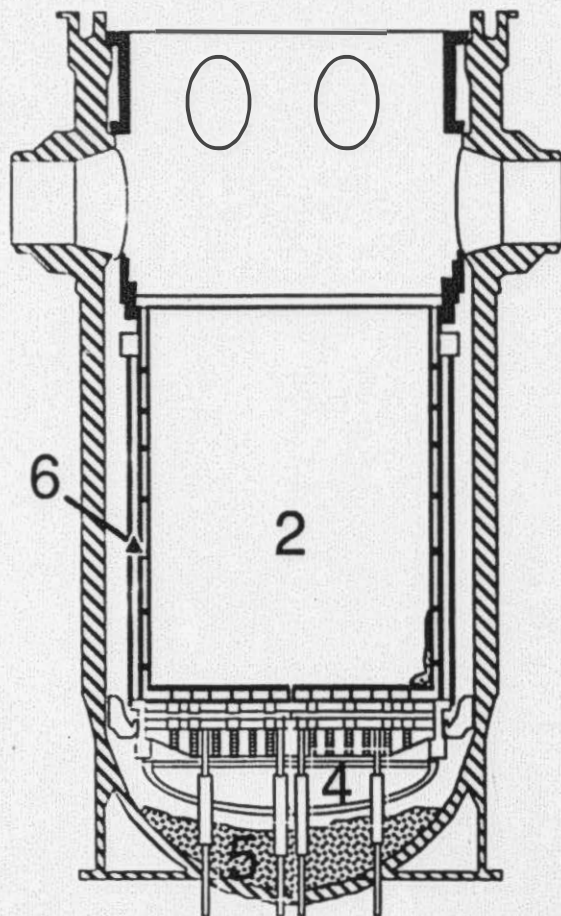
3,800 lb

This includes new material introduced as a result of defueling operations and material due to recently discovered damage to reactor vessel internals.

TOTAL

(Total fuel and structural material to be removed and shipped in canisters)

296,900 lb

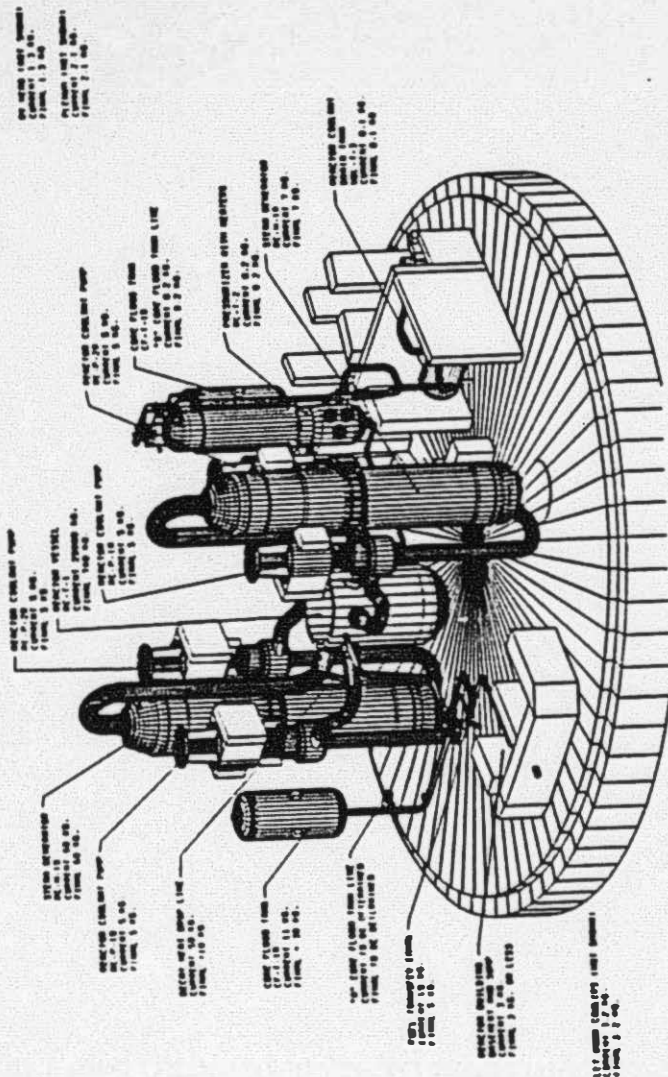


CURRENT DISTRIBUTION

REGION	DESCRIPTION	ESTIMATED QUANTITY (lb)
•	Defueled Material	208,400
2	Core Region Resolidified Mass (R-6 Location)	200
4	Lower CSA a) Resolidified Material b) Loose Material (Vacuumable) d) Loose Material (Rods and Rocks)	17,200
5	Lower Head a) Monolith or Fused Material b) Post-Accident Loose Material (Non-Vacuumable) c) Post-Accident Loose Material (Vacuumable) d) Newly Relocated Loose Material (Vacuumable) e) Newly Relocated Rods and Rocks	60,800
6	Core Former Region	9,300
7	Ex-Vessel RCS	1,000

* All of Regions 1 (loose debris) and 3 (intact assemblies) as well as almost all of Region 2 (resolidified mass) have been defueled or relocated.

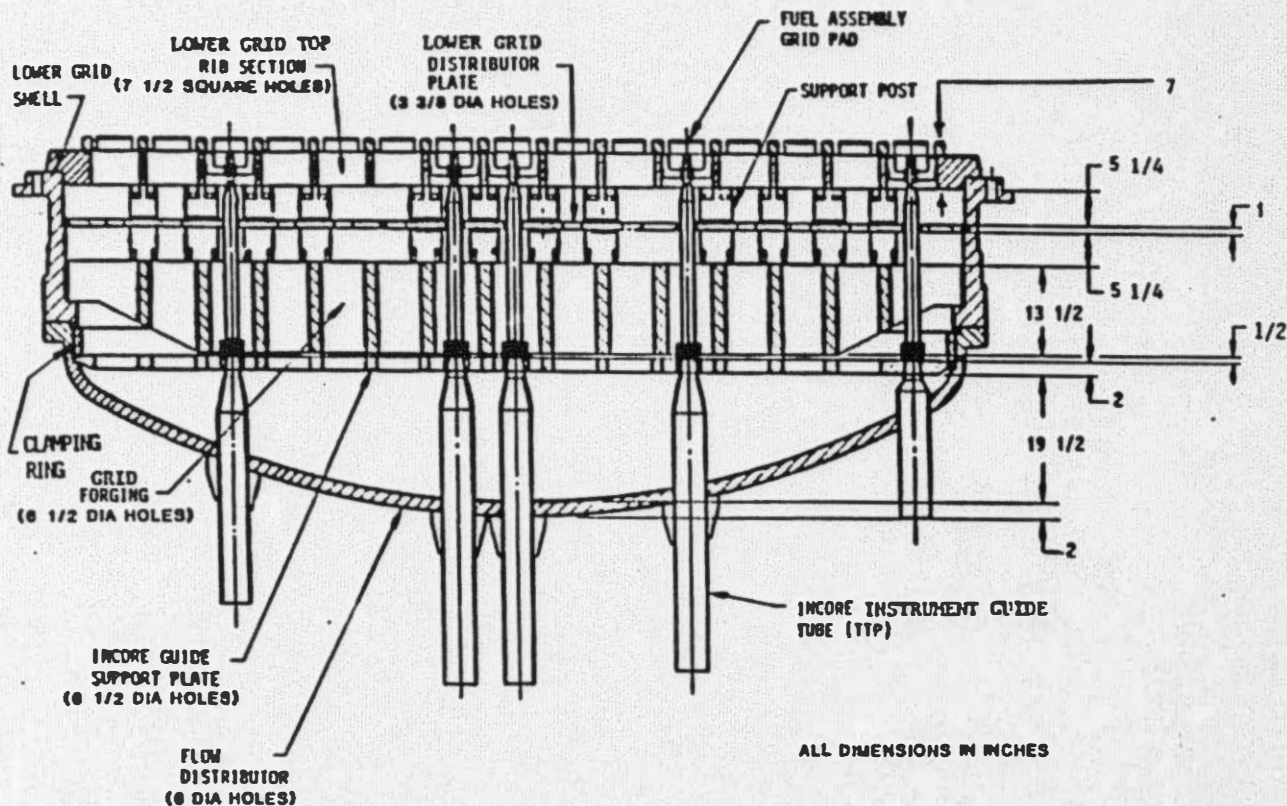
RESIDUAL FUEL INVENTORY



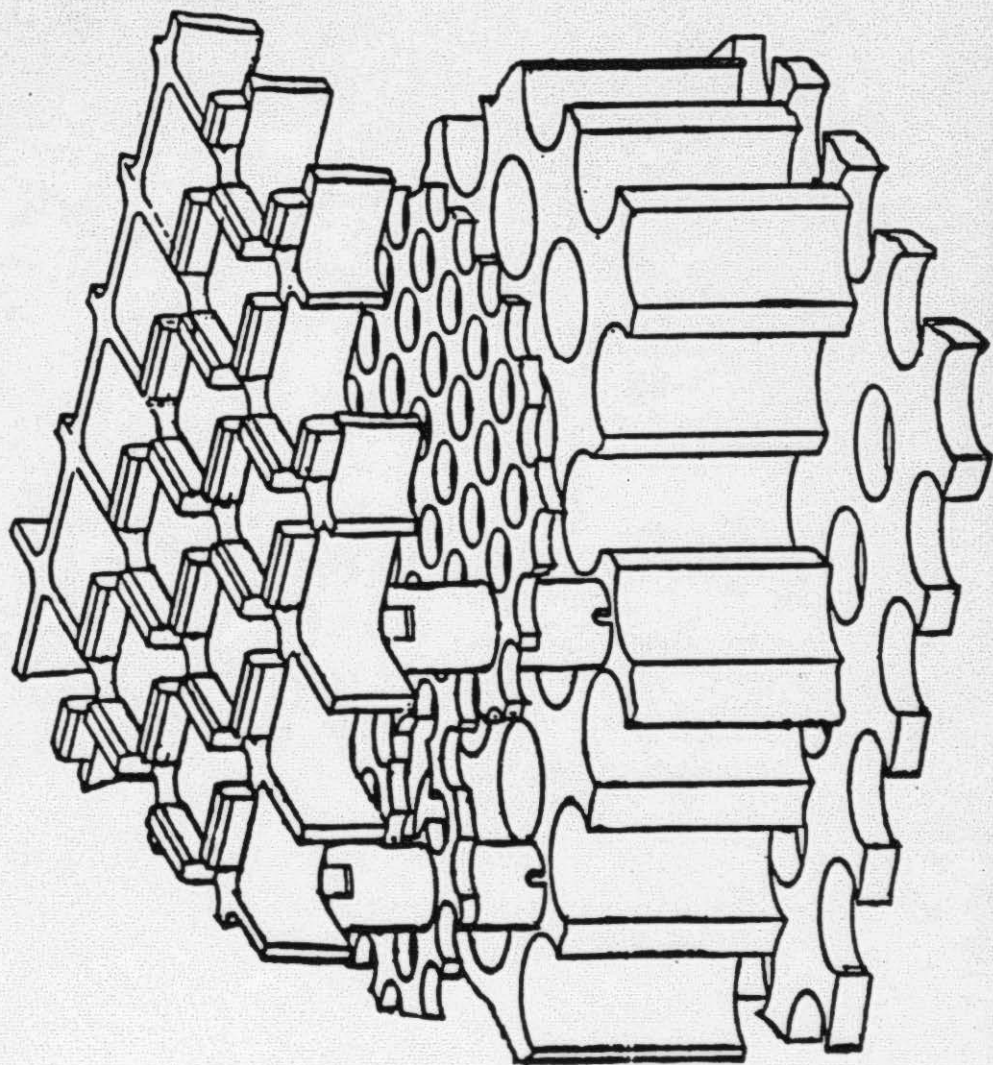
AUX. / FUEL MANOL. BLOO.	CURRENT	FINAL COMB
	+30 MG	+20 MD.

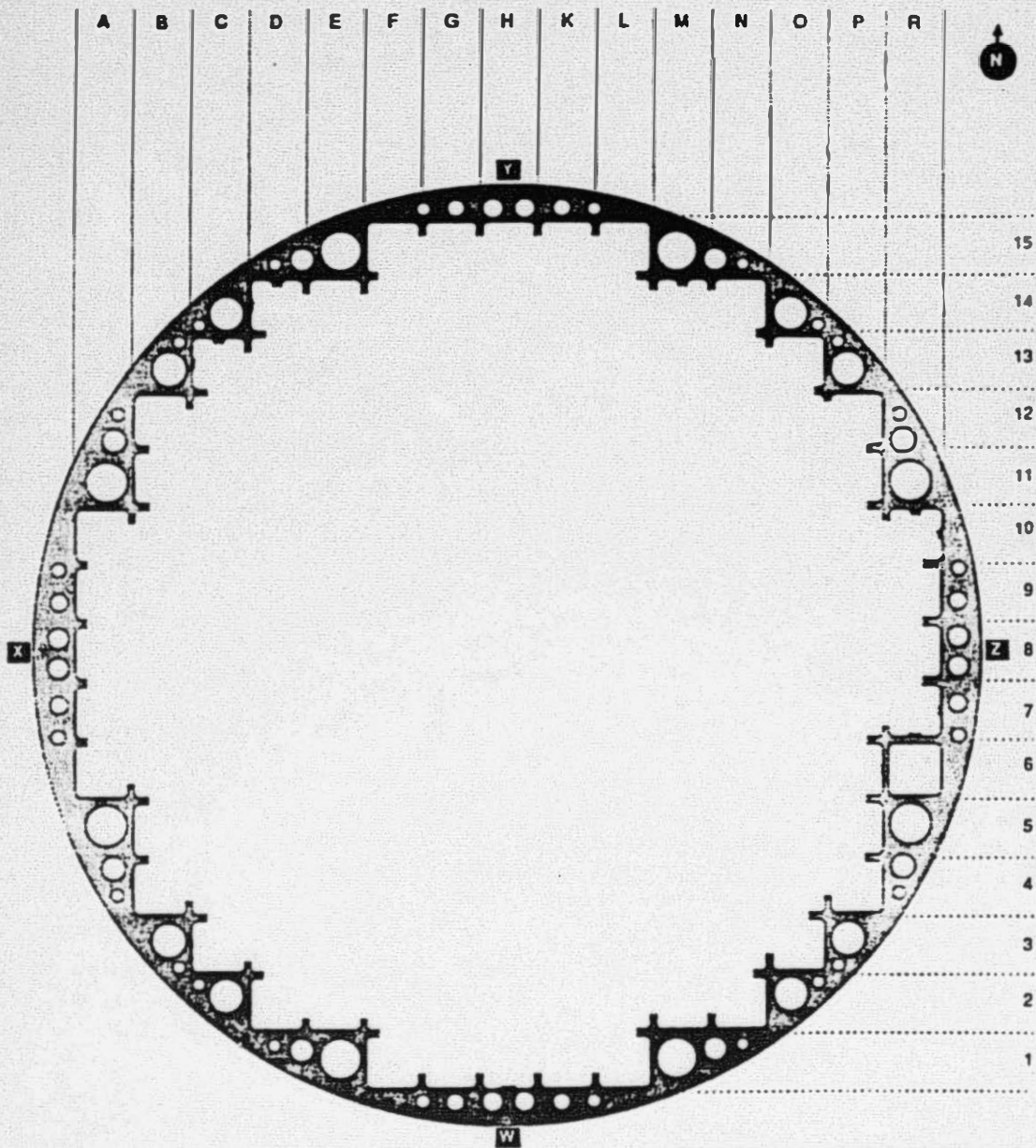
REACTOR BUILDING	
CURRENT INVENTORY	FINAL COAL
29,000 MG.	740 MG.
IN VESSEL	
200 MG.	180 MG.

TMI-2 LOWER CORE SUPPORT ASSEMBLY

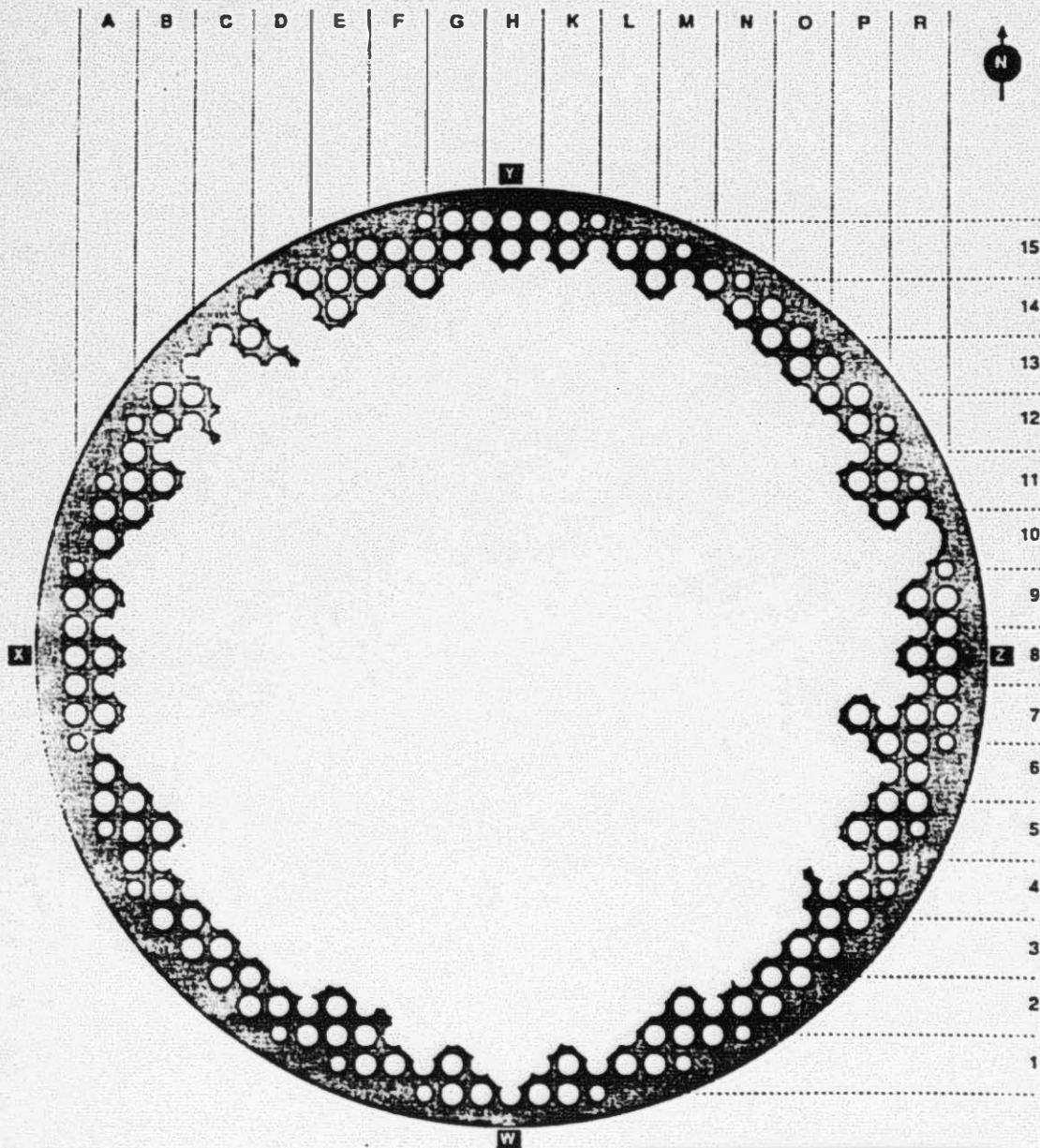


LOWER GRID ASSEMBLY



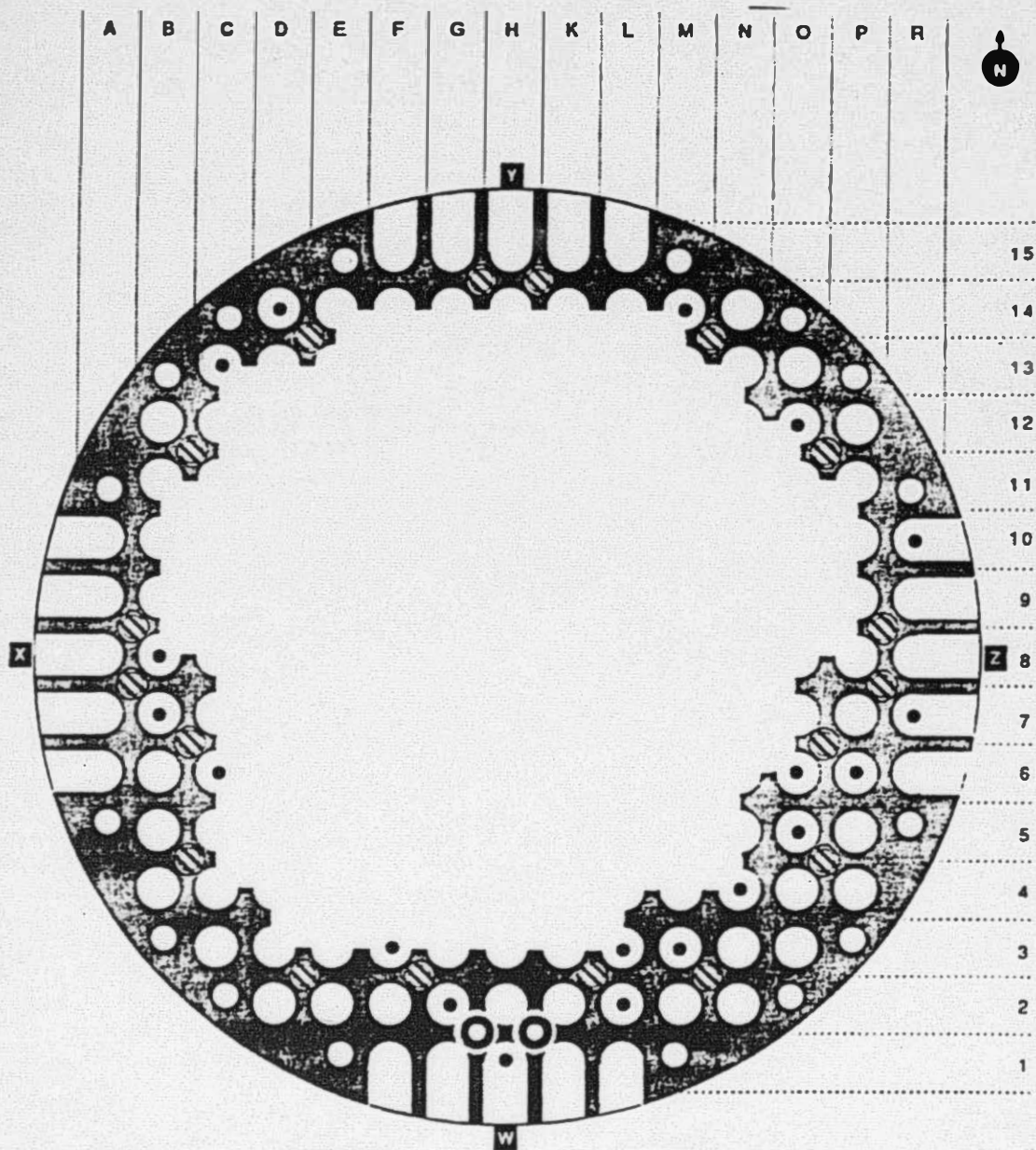


STATUS AS OF 12/2/88



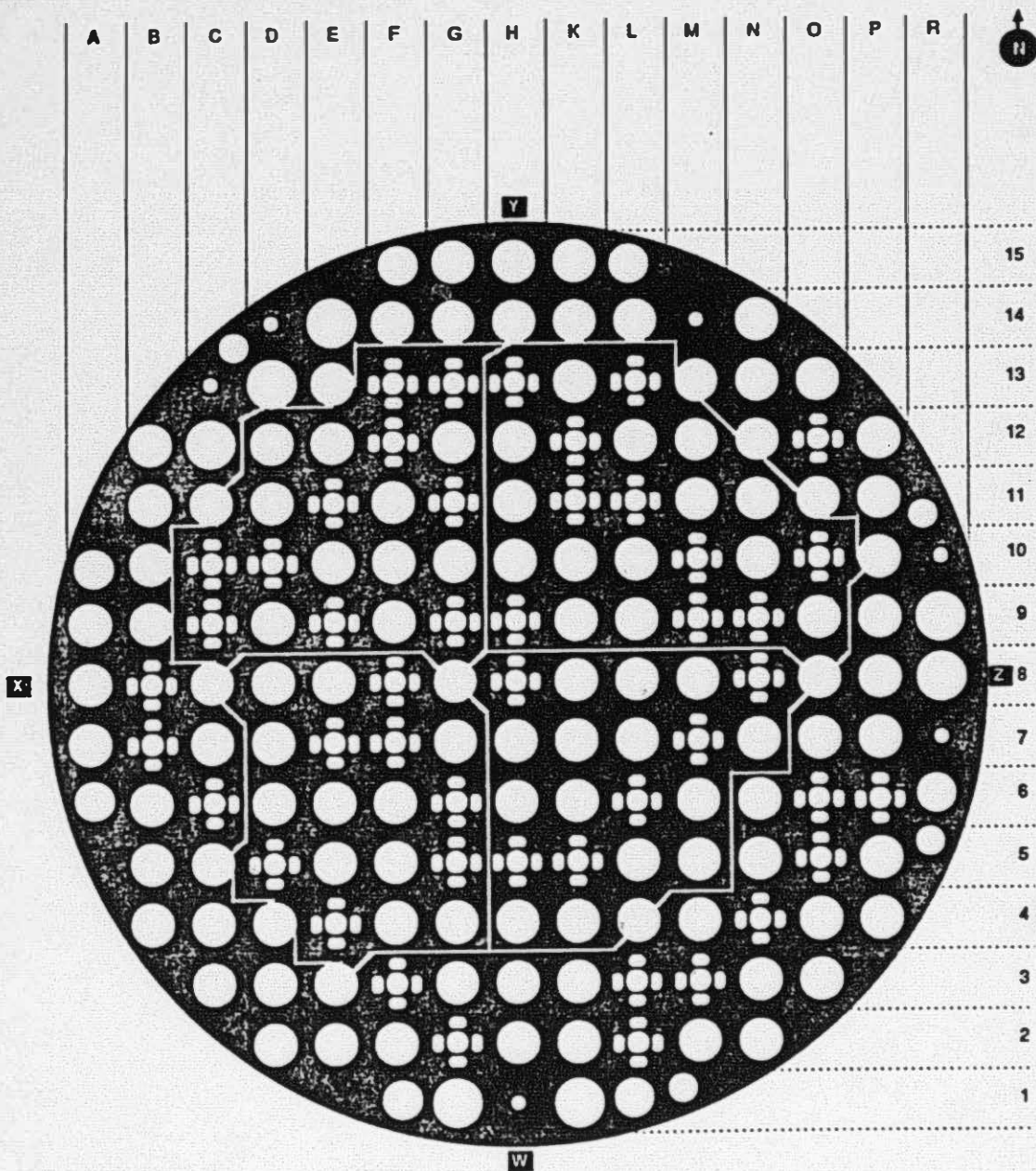
STATUS AS OF 12/2/88

(Incore Guide Tubes And Support Posts Not Shown)

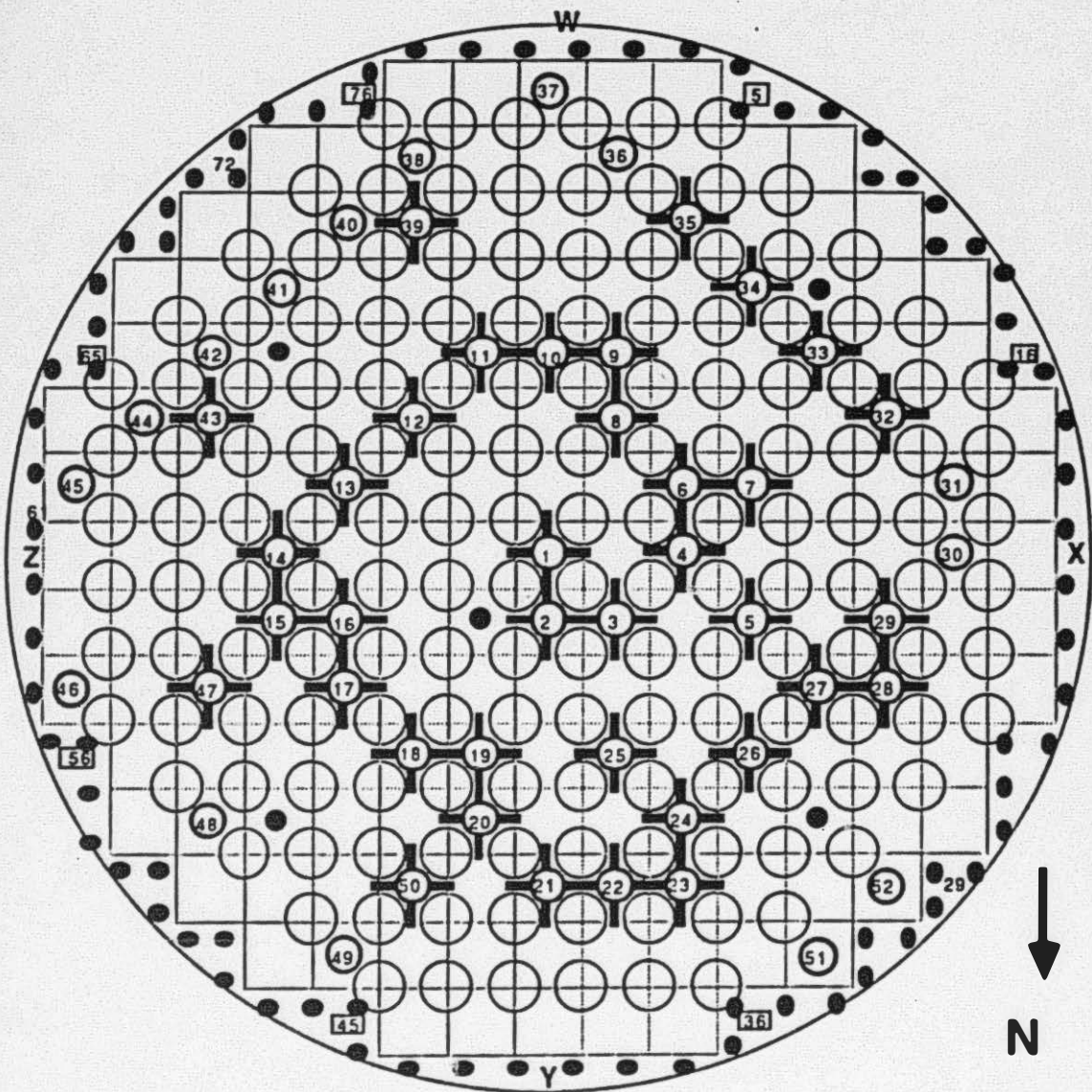


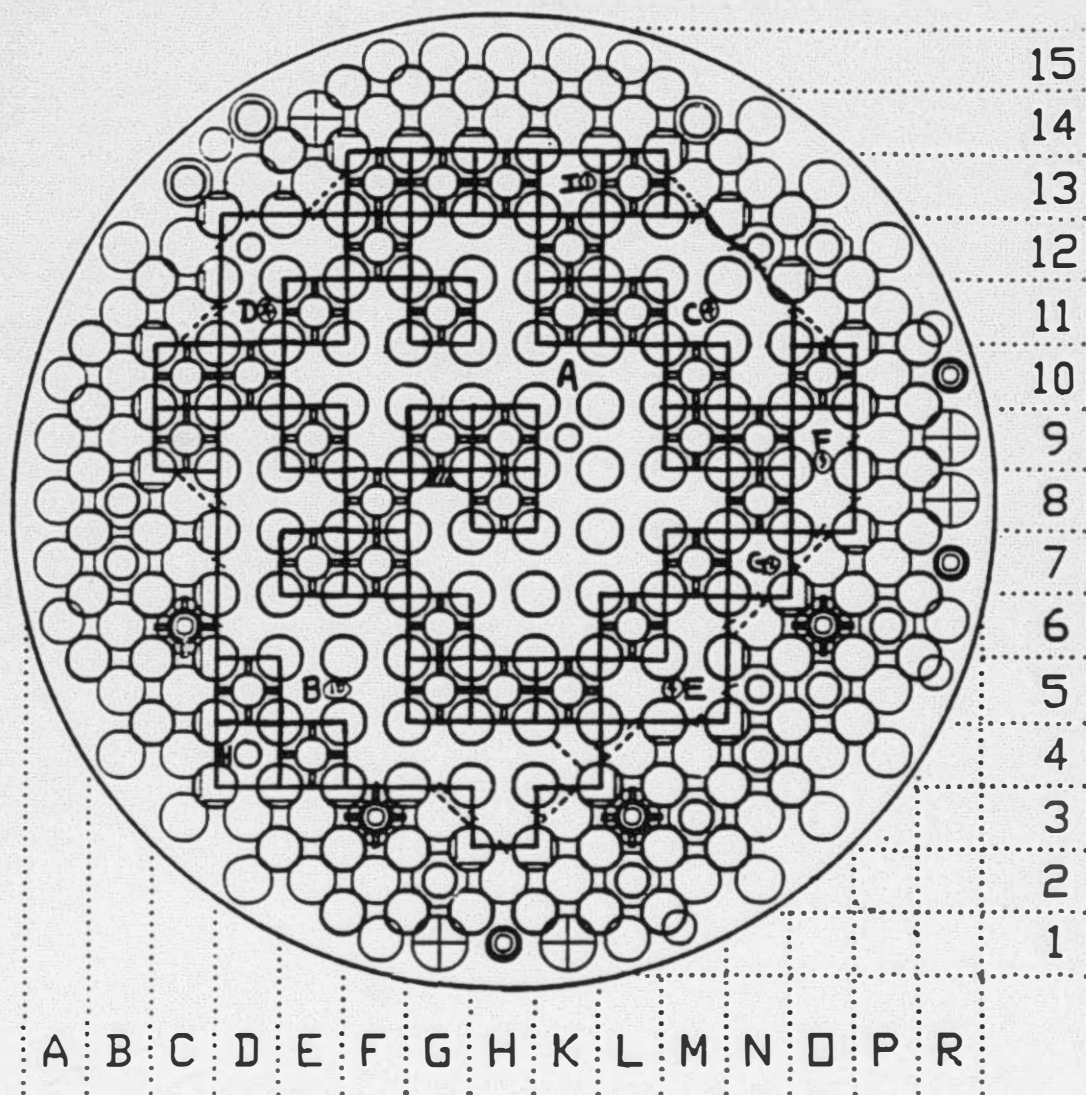
○ Partially Drilled Support Post

STATUS AS OF 12/2/88



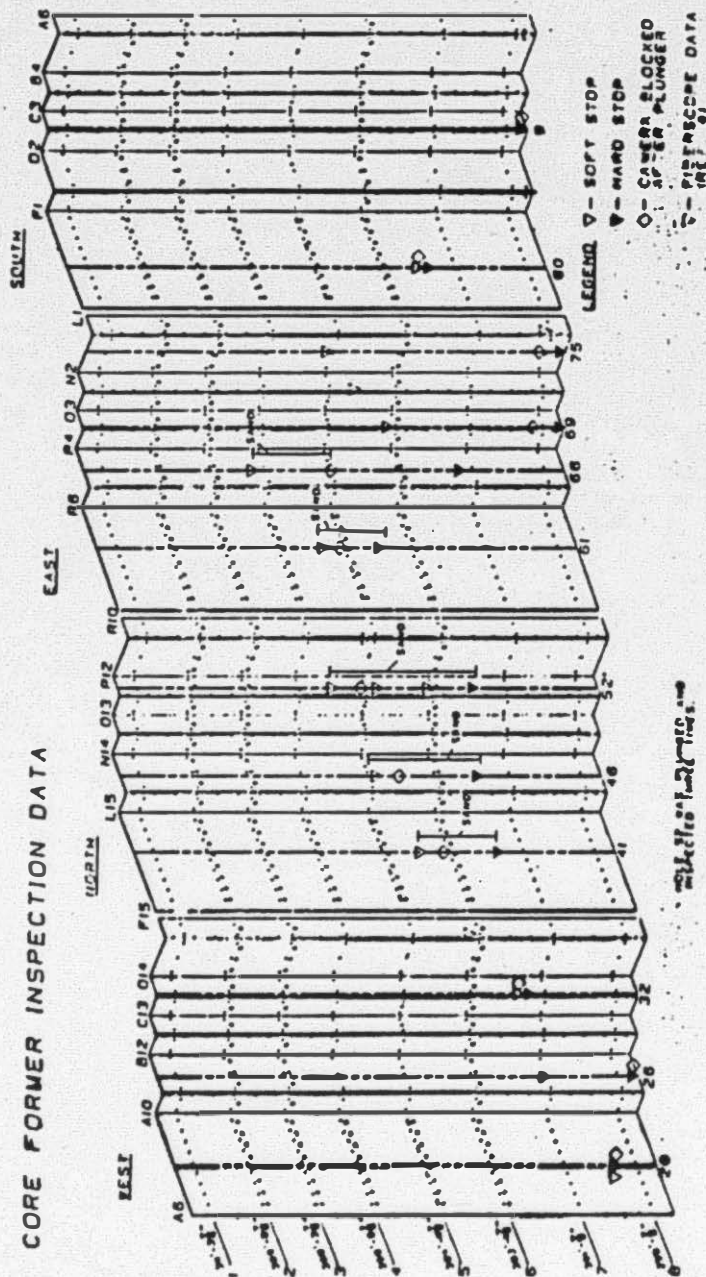
**Incore Guide Support Plate
Approved Cut Pattern**





FLOW DISTRIBUTOR CUTTING

CORE FORMER INSPECTION DATA



UPPER CORE SUPPORT ASSEMBLY DEFUELING

- **CUT BAFFLE PLATES VERTICALLY WITH
PLASMA ARC SYSTEM**
- **UNTHREAD BOLTS SECURING BAFFLE PLATES
TO THE FORMER PLATES**
- **HANDLE BAFFLE PLATES WITHIN THE CORE
REGION OF THE REACTOR VESSEL**

4 CORE FORMER
ELEVATIONS

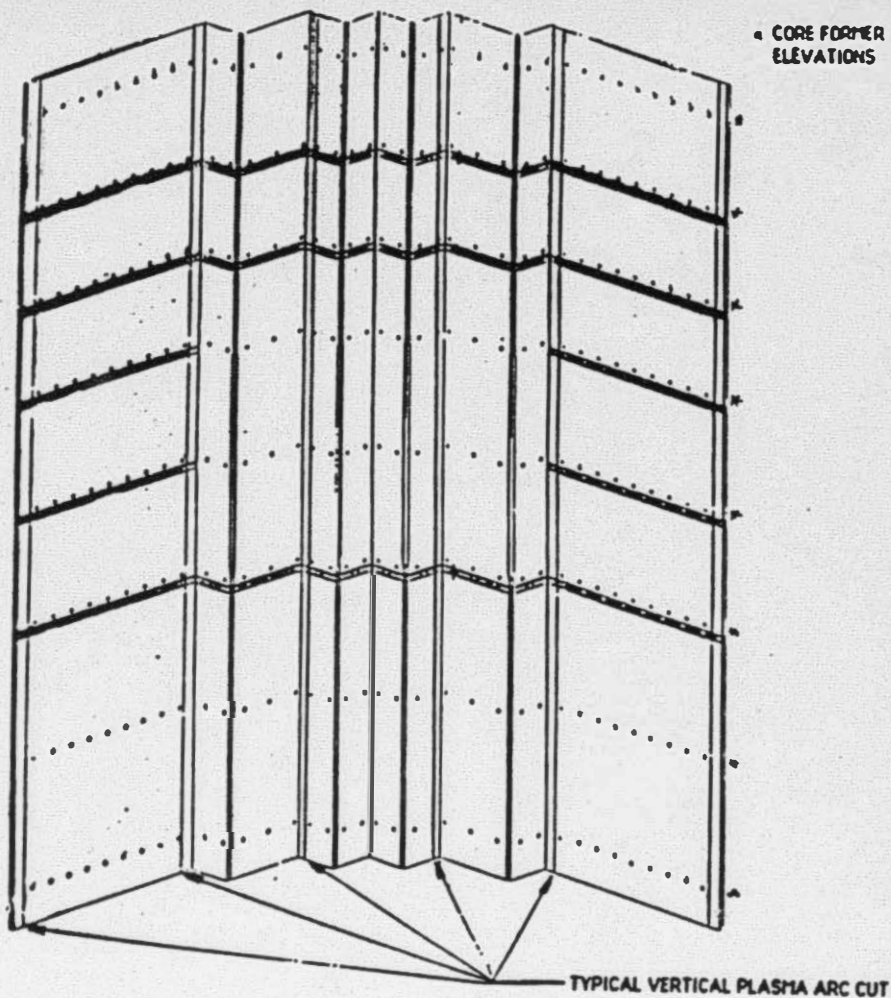


FIGURE 6- BAFFLE PLATE ACCESS OPENINGS

NOTE: HORIZONTAL BLACK BARS DO NOT REPRESENT ANY STRUCTURE, THEY ARE NECESSARY FOR THE CADD SYSTEM TO DEPICT HOLES IN A FLAT PLATE.

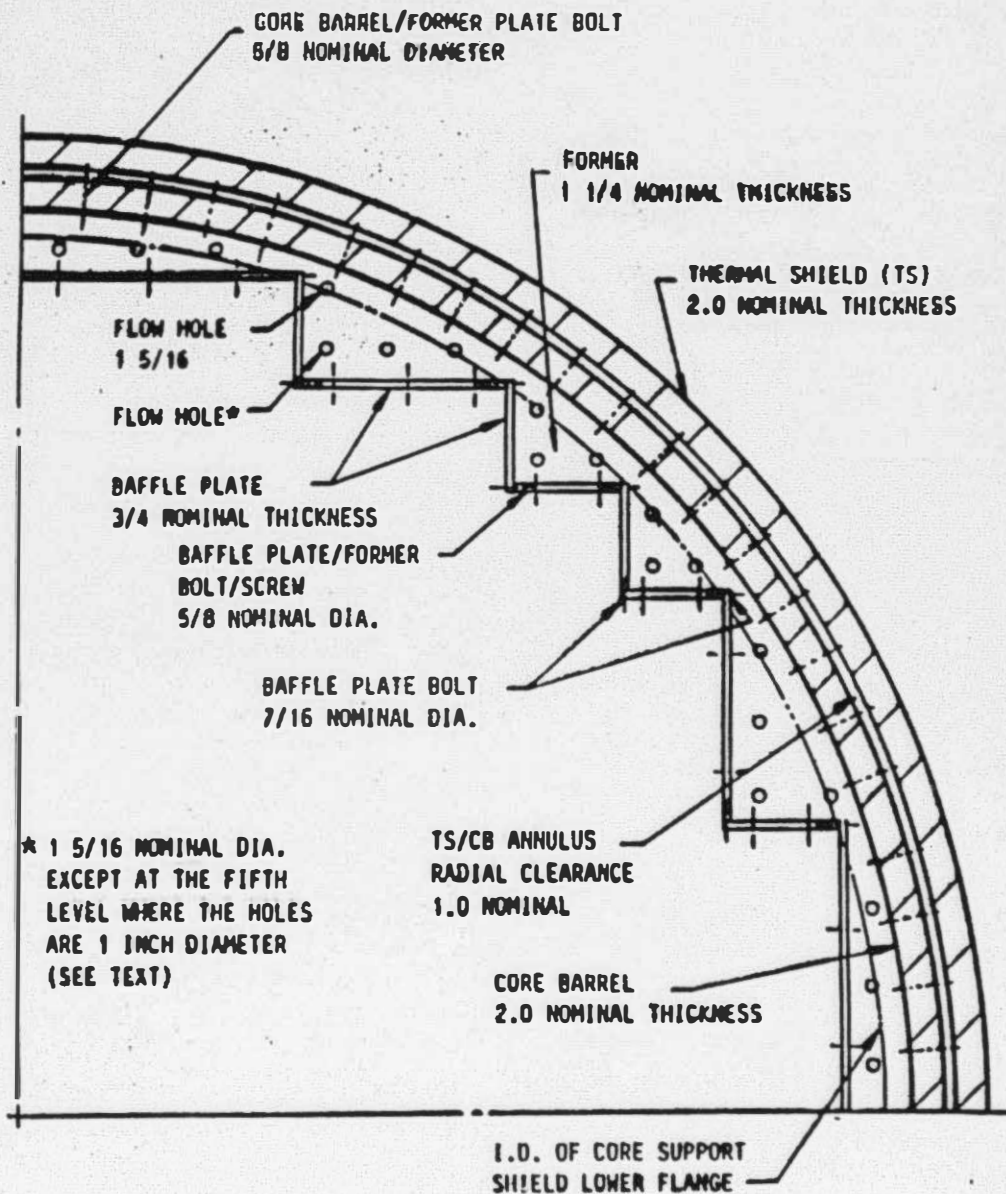
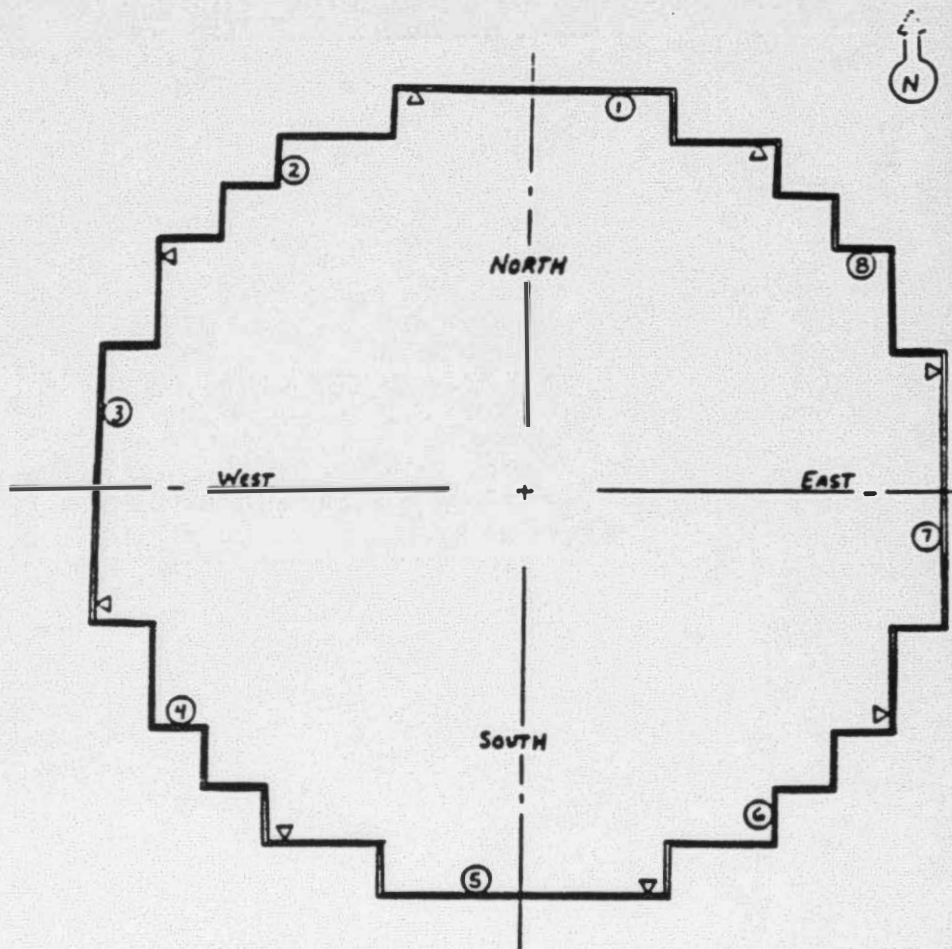


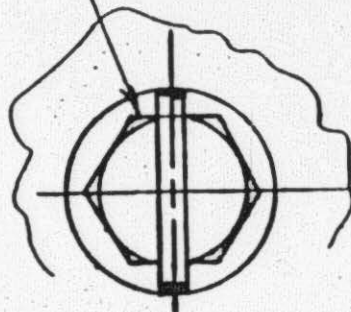
FIGURE 4A- FORMER/BAFFLE PLATE LAYOUT



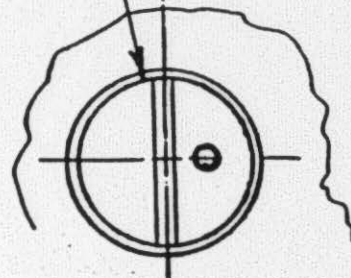
△ — Proposed Cut Locations

Option 1A - Baffle Plate Sections/Cut Locations

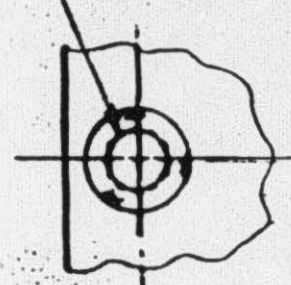
5/8 - 11 x 1 1/2 HEX HD
ST SL BOLT 6 REQD



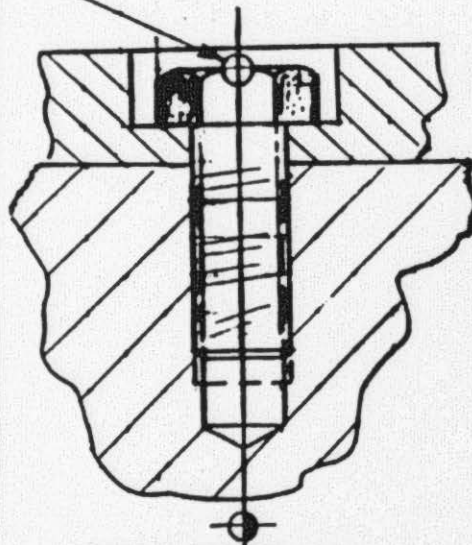
5/8 - 11 x 3/4 ST SL SHLD BOLT
3/4 DIA x 1/2 LG SHLD
4 REQD



7/16 - 14 x 1 12PT HIGH
TORQUE BOLT - ST SL
7 REQD



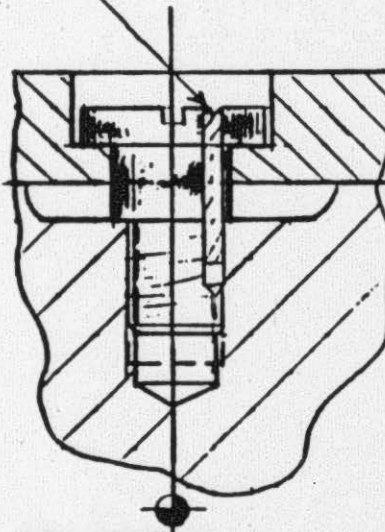
WELD 3/16 PIN



DRILL & TAP 5/8 - 11
C'BORE 1 3/8 DIA x 1/2 DP
6 PLCS

756 TOTAL

WELD 3/16 PIN



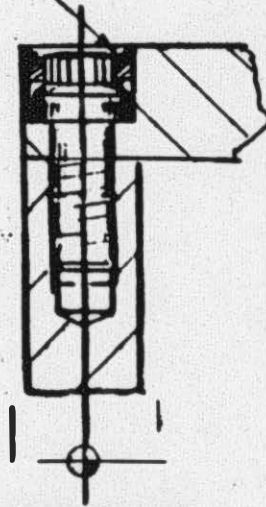
DRILL & TAP 5/8 - 11
C'BORE 1 1/4 DIA x 1/2 DP
4 PLCS

108 TOTAL

FAB LOCKING RING
WELD

.75 PLATE
TYP

.25 GAP

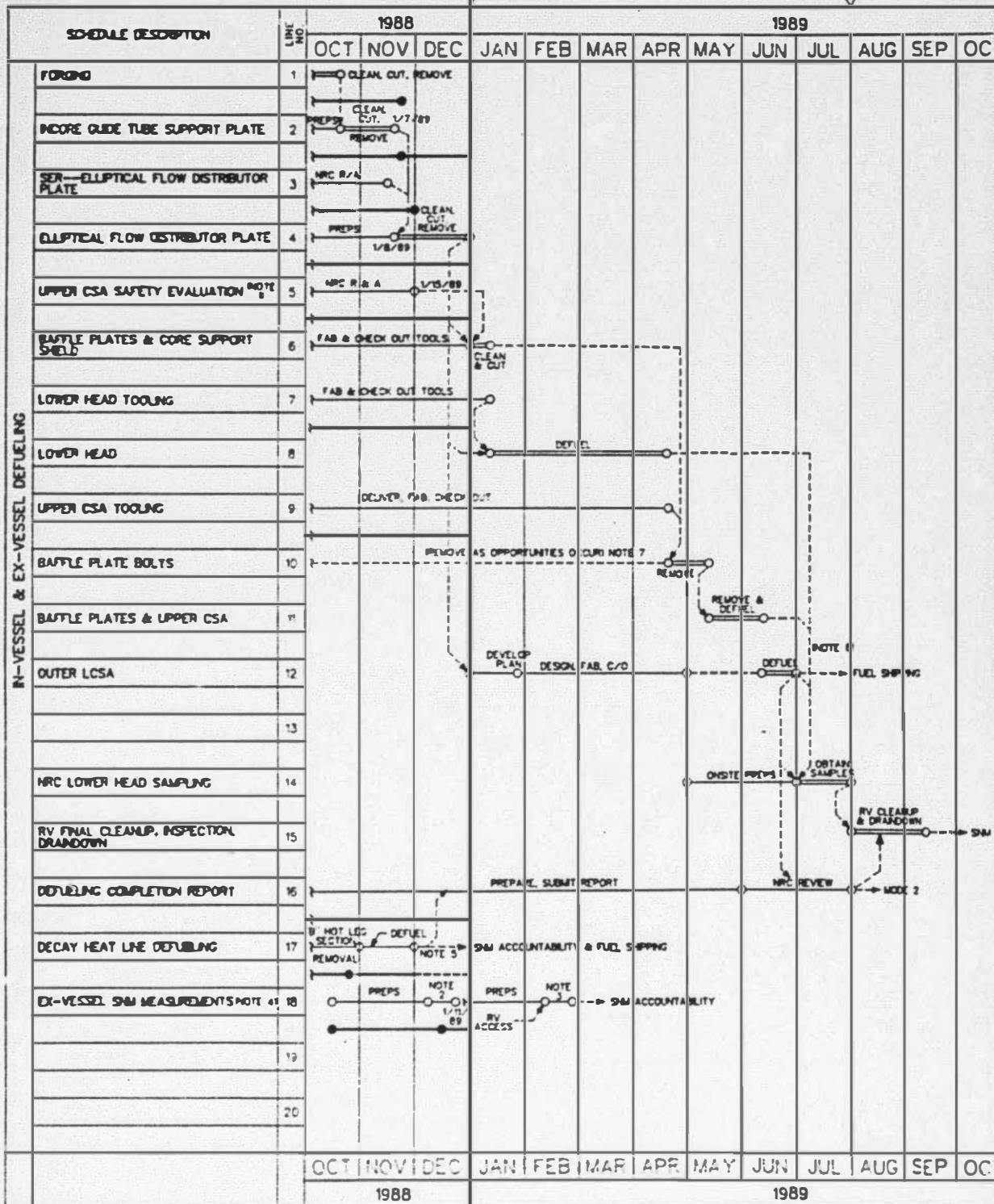


DRILL & TAP 7/16 - 14
C'BORE 11/16 DIA x 1/2 DP
7 PLCS

612 TOTAL

STATUS
THROUGH

MODE 2



TMI-2
DEFUELING
COMPLETION
REPORT

HISTORY OF DEFUELING COMPLETION REPORT

- 04-23-87 - TSCR 53 SUBMITTED FOR NRC REVIEW AND APPROVAL. PROPOSED TO ESTABLISH MODES 1, 2, AND 3.
- 09-18-87 - MEETING WITH NRC TO DISCUSS TSCR 53. (i.e., INCLUDES DEFUELING COMPLETION REPORT.
- 10-26-87 - GPU NUCLEAR SUBMITS REVISION 1 TO TSCR 53 IN RESPONSE TO NRC COMMENTS. TECH. SPEC. 1.3, "MODE," IS REVISED TO REQUIRE A DEFUELING COMPLETION REPORT BE SUBMITTED WITHIN 30 DAYS PRIOR TO MODE CHANGE.
- 01-13-88 - NRC ISSUED THE NSHC FOR TSCR 53 WHICH RESTATES REQUIREMENT FOR 30-DAY REPORT.
- 05-27-88 - NRC LICENSE AMENDMENT NO. 30 ISSUED APPROVING. NRC REQUIRES REPORT TO BE SUBMITTED WITHIN 60 DAYS PRIOR TO MODE CHANGE.

DEFUELING COMPLETION REPORT

- CRITICALITY ANALYSIS ADDRESSING EACH SEPARATE QUANTITY OF RESIDUAL FUEL. INCLUDES:
 - Estimate of the quantity of residual fuel
 - Its location and dispersion with the location
 - Physical form
 - Mobility
 - Presence of any moderator or reflecting material
- CRITICALITY ANALYSIS MUST DEMONSTRATE THAT DEFUELING HAS PROGRESSED SUFFICIENTLY THAT INADVERTENT CRITICALITY IS PRECLUDED (i.e., LONG-TERM SUBCRITICALITY IS ASSURED).
- REPORT DOES NOT REQUIRE NRC APPROVAL.

TMI-2

ACCIDENT GENERATED WATER
DISPOSAL

SYSTEM UPDATE

WATER DISPOSAL STATUS

DIVISION OF RESPONSIBILITY

PACIFIC NUCLEAR

- Design
- Fabrication
- Install
- Test
- Operate
- Remove Equipment

GPUN

- Provide water complying with influent criteria
- Ship solid waste
- Provide Rad Con services
- Chemistry Support
- Safety evaluation, licensing

SCHEDULE

- SHOP TESTS JANUARY 16
- MOBILIZE TO LANCASTER JANUARY 23
- MECHANICAL/ELECTRICAL TEST FEBRUARY 6-15
- MOBILIZE AT TMI FEBRUARY 21
- COMPLETE ASSEMBLY/ERECTION MARCH 30
- ACCEPTANCE TESTING APRIL 3
- AVAILABLE FOR OPERATIONS APRIL 10

VENDOR'S TEST PROGRAM

- SHOP TESTING

DEMONSTRATES OPERABILITY (INCLUDING DF)
OF EVAPORATOR AND VAPORIZER

- MECHANICAL/ELECTRICAL TEST

DEMONSTRATES PROPER FUNCTION OF
INSTRUMENTS, INTERLOCKS, MECHANICAL
FIT-UP OF ALL VENDOR SUPPLIED EQUIPMENT

- ACCEPTANCE TESTING

OPERABILITY OF ALL EQUIPMENT USING
NON-RADIOACTIVE WATER

CURRENT STATUS

VENDOR

- TEST PLANNED FOR WEEK OF JANUARY 16, 1989 TO OPERATE EVAPORATOR AND VAPORIZER. KEY PURPOSE IS TO DETERMINE BORON DF.

SITE

- CONCRETE PAD AND PRIMARY ELECTRICAL COMPLETE
- FLUID SYSTEM TIE-INS AND ASSOCIATED ELECTRICAL EQUIPMENT BEING WORKED

LICENSING

- ASLB PROCESS UNDERWAY
- PADER PERMIT PENDING

PROCESSED WATER DISPOSAL SYSTEM

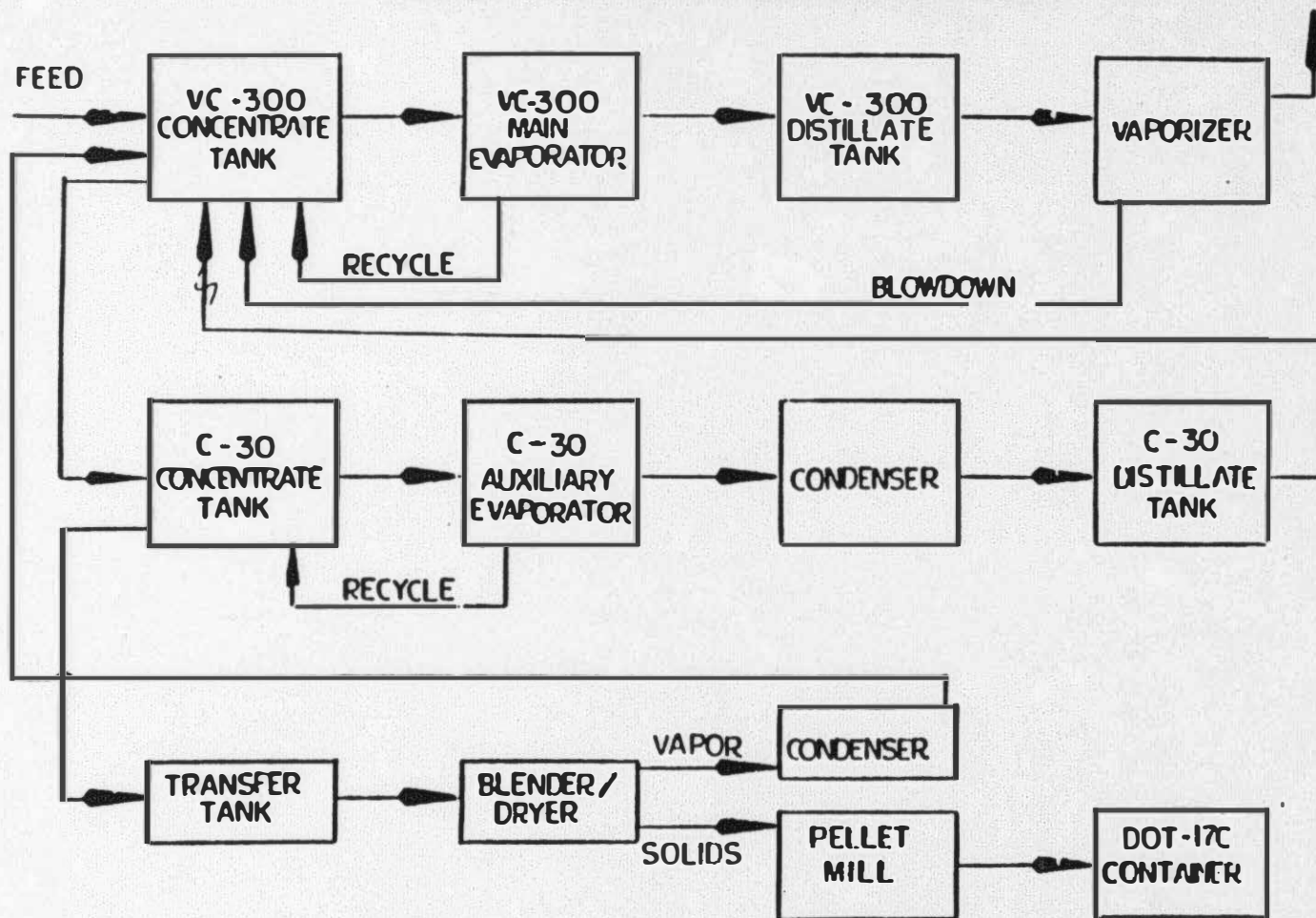


Figure 1: PROCESS FLOW BLOCK DIAGRAM

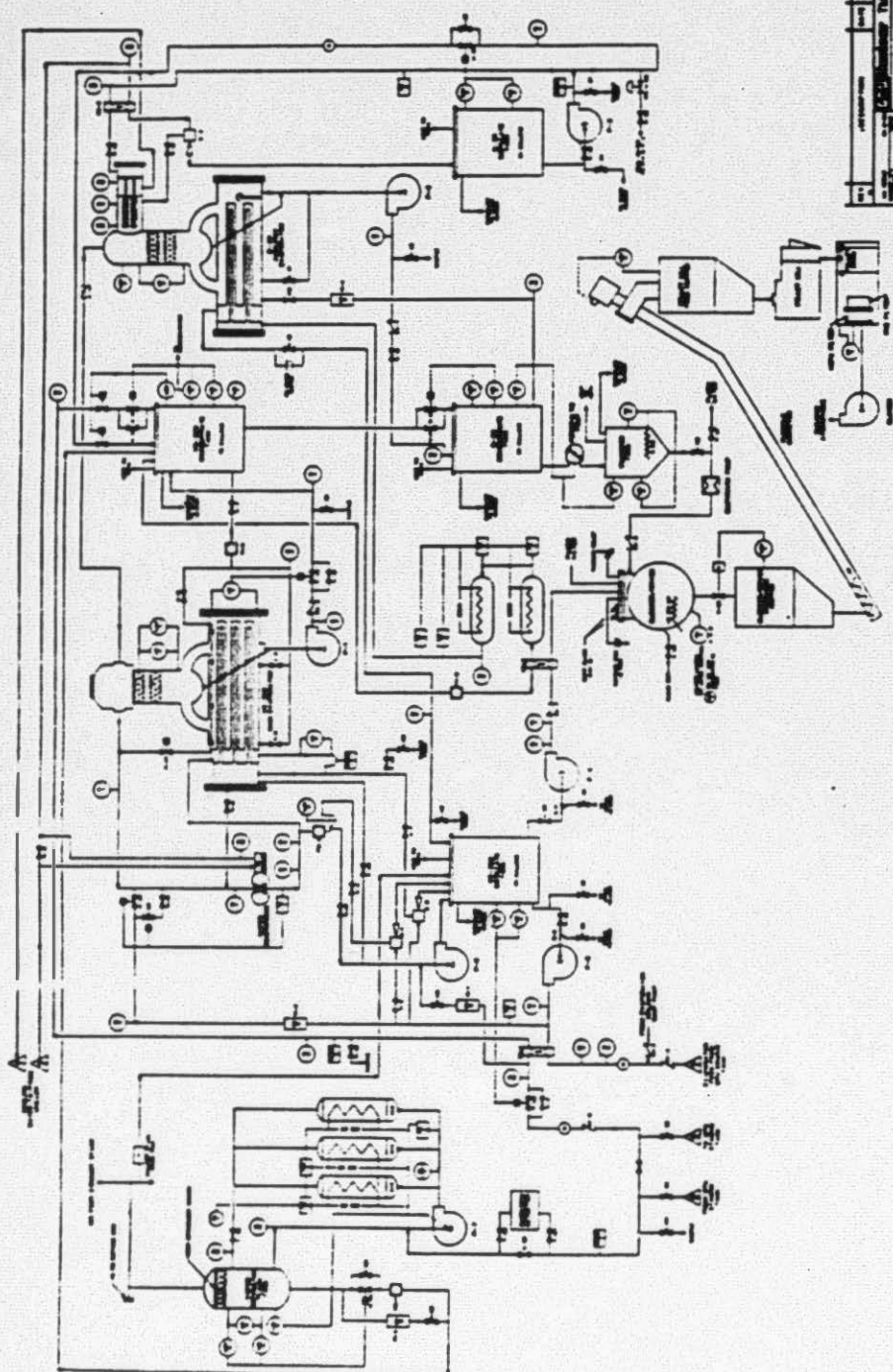
PARAMETER	VALUE ($\mu\text{C}/\text{ml}$)
TRITIUM	1.3E-1
CESIUM 137	3.7E-5
STRONTIUM 90	1.1E-4
CARBON 14	1.0E-4
ANTIMONY 125	2.3E-6
TECHNETIUM 99	1.0E-6
CESIUM 134	8.8E-7
COBALT 60	4.8E-7
IRON 55	4.8E-7
BIRON (PPM)	3000
SODIUM (PPM)	700
IODINE 129	<6.0E-7
CERIUM 144	<1.8E-8
MANGANESE 54	<4.0E-8
COBALT 58	<4.0E-8
NICKEL 63	<6.0E-7
ZINC 65	<9.8E-8
RUTHENIUM 106	<3.3E-7
SILVER 110M	<5.6E-8
PROMETHIUM 147	<4.8E-6
EUROPIUM 152	<3.8E-10
EUROPIUM 154	<4.4E-8
EUROPIUM 155	<1.1E-7
URANIUM 234	<1.0E-8
URANIUM 235	<1.2E-8
URANIUM 238	<1.2E-8
PLUTONIUM 238	<1.2E-8
PLUTONIUM 239	<1.4E-8
PLUTONIUM 240	<1.4E-8
PLUTONIUM 241	<6.5E-7
AMERICIUM 241	<1.2E-8
CURIUM 242	<1.0E-7

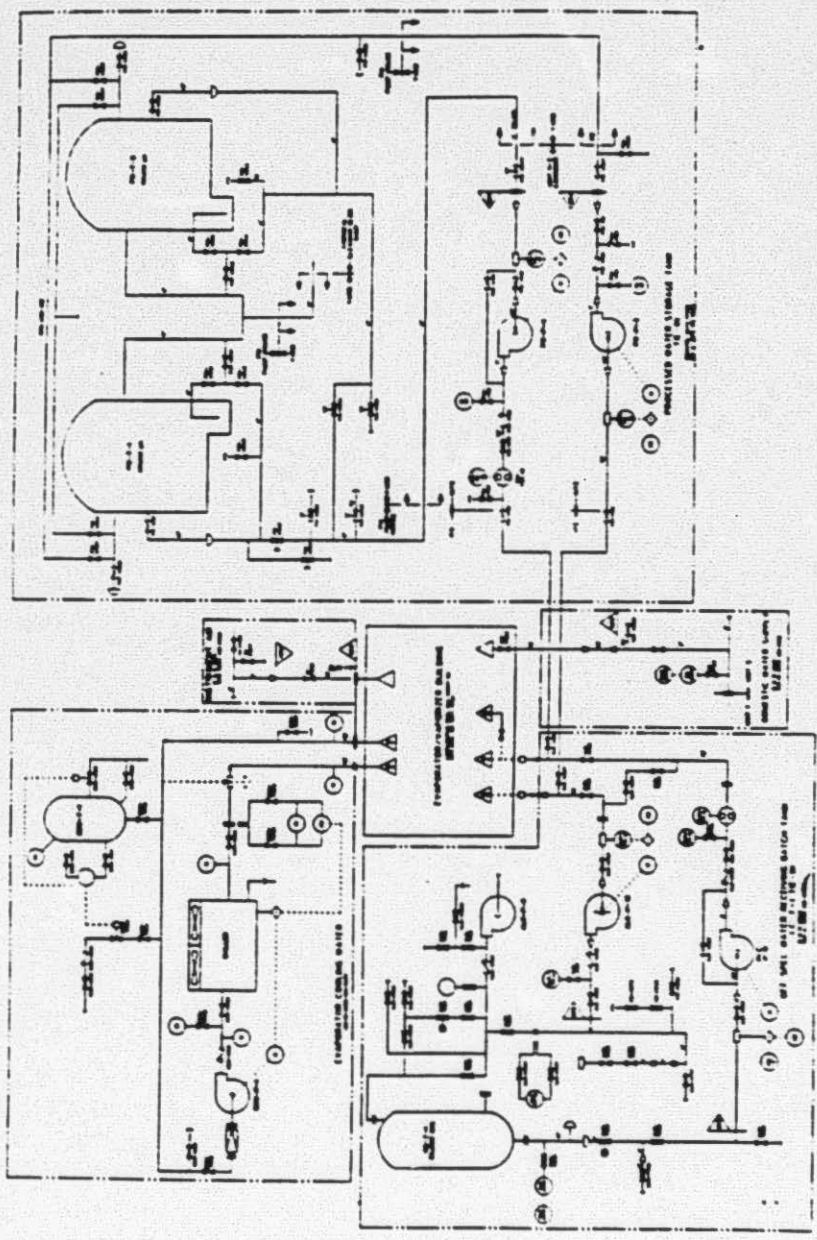
PWST-2

PARAMETER	VALUE (uc/ml)
TRITIUM	1.3E-1
CESIUM 137	1.2E-5
STRONTIUM 90	2.0E-5
CARBON 14	
ANTIMONY 125	6.4E-7
TECHNETIUM 99	
CESIUM 134	2.0E-7
COBALT 60	3.3E-7
IRON 55	
BORON (PPM)	2028
SODIUM (PPM)	1.6
IODINE 129	
CERIUM 144	<7.0E-7
MANGANESE 54	
COBALT 58	<3.0E-8
NICKEL 63	
ZINC 65	
RUTHENIUM 106	<6.3E-7
SILVER 110M	<2.6E-7
PROTHIUM 147	
EUROPIUM 152	
EUROPIUM 154	
EUROPIUM 155	
URANIUM 234	
URANIUM 235	
URANIUM 238	
PLUTONIUM 238	
PLUTONIUM 239	
PLUTONIUM 240	
PLUTONIUM 241	
AMERICIUM 241	
CURTIUM 242	

PWST-2

PARAMETER	VALUE
BORON (PPM)	2028
SODIUM (PPM)	1.6
pH	5.46
TURBIDITY (NTU)	0.23
T.O.C. (PPM)	
CONDUCTIVITY (umho/cm)	9.9
SILICA (SiO2) (PPM)	
CHLORIDE (PPM)	<0.007
NITRATE (PPM)	0.06
PHOSPHATE (PPM)	<0.095
SULFATE (PPM)	0.17
TOTAL IRON (PPM)	





POST DEFUELING SURVEY
FOR
SPECIAL NUCLEAR MATERIAL
ACCOUNTABILITY

BACKGROUND

CURRENT CORE STATE PRECLUDES NORMAL FUEL
ACCOUNTABILITY PROCESS AS REQUIRED BY 10 CFR
70.51(d) AND 10 CFR 70.53

LACK OF SOPHISTICATED ONSITE HOT CELL FACILITIES
PRECLUDE ACCURATE ACCOUNTABILITY OF FUEL
SHIPPED TO DOE/INEL IN FUEL CANISTERS

IN OCTOBER, 1985 THE NRC ISSUED GPU NUCLEAR AN
EXEMPTION FROM THE REQUIREMENTS OF 70.51(d)
AND 70.53 AND NOTED THE GPU NUCLEAR/DOE
AGREEMENT ON THE USE OF AN ANNOTATED
DOE/NRC FORM 741 FOR USE IN MEETING THE
TRANSFER REQUIREMENTS OF 10 CFR 70.54

THEREFORE

GPUN MUST ANNOTATE A DOE/NRC FORM 741 FOR
EACH CANISTER SHIPMENT TO INDICATE THAT
QUANTIFICATION OF SNM ON A PER CANISTER BASIS
IS NOT POSSIBLE AND
A FINAL ACCOUNTING OF SNM WILL BE PERFORMED
AT THE COMPLETION OF DEFUELING AT WHICH TIME A
POST DEFUELING SURVEY WILL QUANTIFY THE
RESIDUAL FUEL INVENTORY

FINAL SNM ACCOUNTABILITY

CURRENT INVENTORY (DECAY CORRECTED)

- SNM SHIPPED AS RADWASTE AND SAMPLES
- FINAL PLANT INVENTORY AS DETERMINED
BY THE POST DEFUELING SURVEY

= SNM SHIPPED TO DOE/INEL IN CANISTERS

SNM MEASUREMENT TECHNIQUES

GAMMA SPECTROSCOPY

Sodium Iodide
High Purity Germanium

NEUTRON DETECTION

Metal Foil Activation
BF3 Neutron Detectors
Active Neutron Interrogation

ALPHA DETECTION

Cylindrical Gas Proportional Detector

VISUAL INSPECTION

Remote inspection via boroscope, fiberscope, Rees
Camera and normal television camera

SAMPLING

Verification of ratio of tracer isotopes to fuel Uranium
content ratio

POST DEFUELING SURVEY SNM MEASUREMENTS

SNM measurements conducted as part of the Post Defueling Survey will be performed to quantify for final SNM accountability the residual SNM inventory with the facility in the draindown configuration.

The results of some measurements performed for the purpose of fuel characterization and/or for the Defueling Completion Report may be used in conjunction with the Post Defueling Survey Measurements

The results of the Post Defueling Survey will be documented in the PDSR Notebook. Two(2) of the Post Defueling Survey Reports ,which are part of the PDSR Notebook, have been issued.

CURRENT STATUS

1. The TMI-2 SNM Accountability Plan has been reviewed , approved and Issued by GPU Nuclear.
2. An Integrated residual fuel measurement plan , which describes the status of Individual SNM measurements, is prepared and issued on a monthly basis.
3. Measurement of residual fuel quantities in the Auxilary and the Fuel Handling buildings is In process. Nine (9) of fourty four (44) cubicles have been completed.
4. Measurement of the residual fuel quantities in the Reactor building locations and Reactor Coolant System components outside of the Reactor Vessel is nearing completion.
5. SNM measurements required to support submission of the Defueling Completion Report will be completed by the end of the second quarter of 1989.
6. SNM measurements required to support the completion of the Post Defueling Survey and final SNM accountability will be finished after the end of fuel shipping activities.

QUALITY ASSURANCE FOR SNM ACCOUNTABILITY

SNM ACCOUNTABILITY PLAN

Based upon QA/QC input, review and comment

MEASUREMENTS

QA approved measurement procedures and
equipment calibration procedures

ANALYSES

All analyses are documented in accordance with the
TMI-2 Engineering Calculations procedure ,which
includes a requirement for independent verification

ALARA FOR SNM ACCOUNTABILITY

Limit SNM measurements to only those essential

Use previous fuel characterization measurements
where possible

Perform each measurement in the most dose
conservative manner

TMI - 2

RADIOLOGICAL CONTROLS

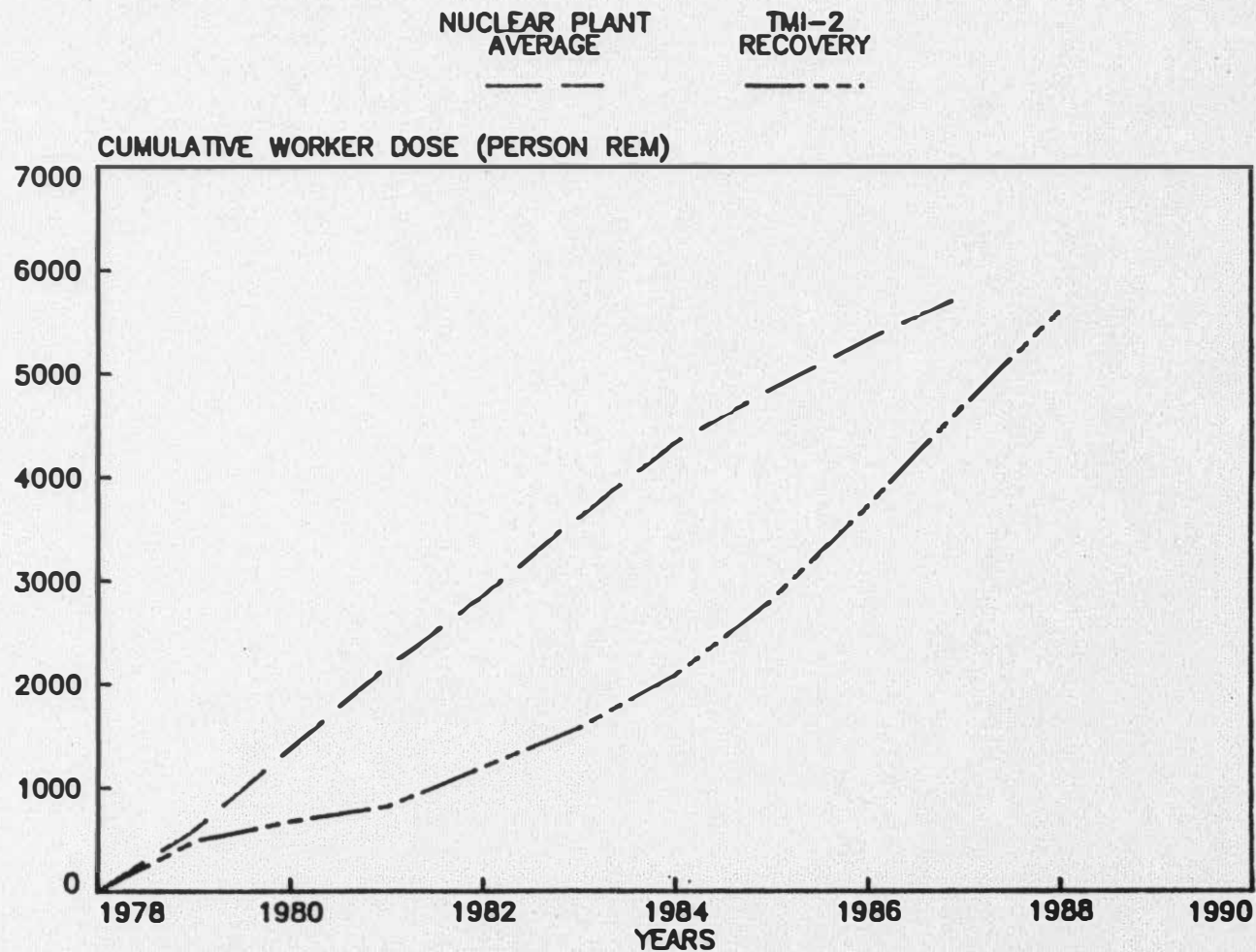
UPDATE

TMI-2 WORKER EXPOSURE

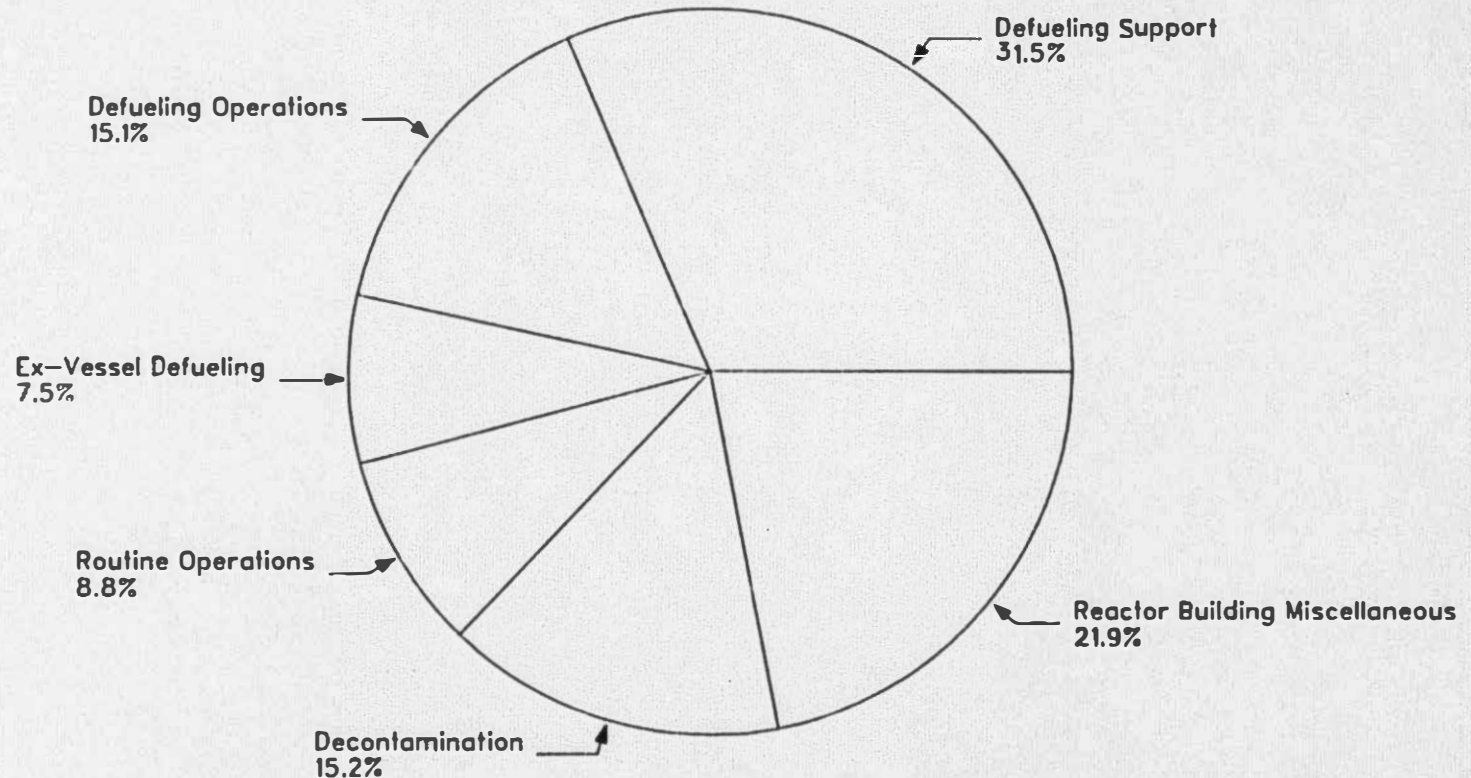
YEAR	PERSON-REM	MAXIMUM WORKER
		WHOLE BODY EXPOSURE
1979*	418	4.5
1980	193	2.1
1981	138	2.0
1982	384	3.0
1983	373	2.7
1984	514	3.7
1985	722	3.5
1986	907	3.4
1987	975	3.5
1988 (to 12/31/88)	917	3.6
TOTAL	5541	

*From March 28, 1979 through December 31, 1979

COMPARISON OF WORKER DOSES



TMI-2 Dose for Current Major Activities (1/1/87 - 12/31/88)



1988 TMI-2 EXPOSURE SUMMARY

(IN PERSON-REM)

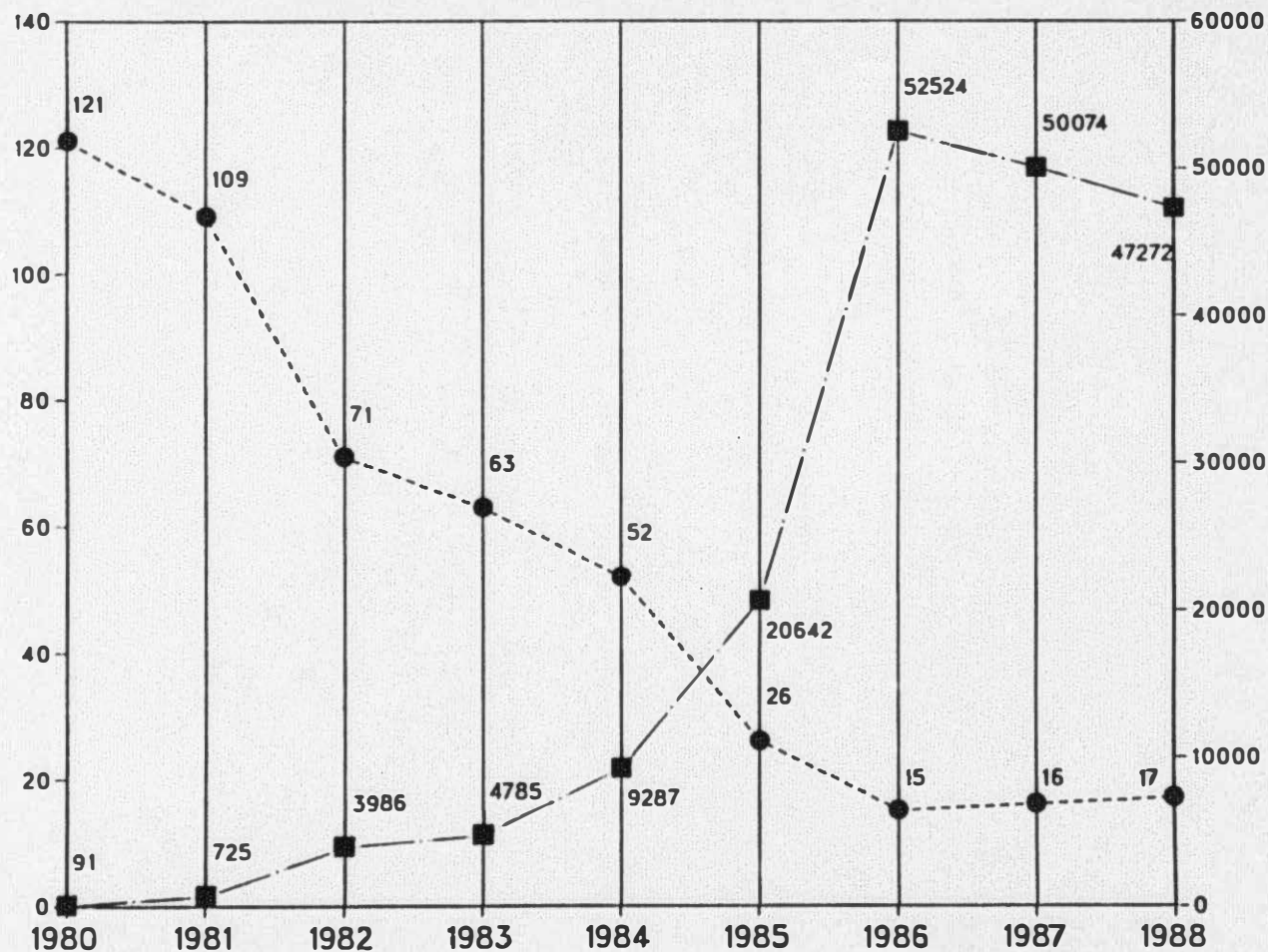
	<u>1988 ESTIMATE</u>	<u>1988 ACTUAL</u>
DEFUELING OPERATIONS (IN-VESSEL ONLY)	253	126
DEFUELING SUPPORT	198	290
BALANCE-OF-PLANT DECON	77	105
BALANCE-OF-PLANT OPERATIONS	74	92
MISC. ACTIVITIES (ROBOTICS,DESLUDGING,RB CRANE OPS,ETC.)	220	227
EX-VESSEL DEFUELING	<u>172</u>	<u>77</u>
TOTAL	994	917

NOTE: ALL PERSON-REM TOTALS ARE CORRECTED TLD VALUES

AVERAGE RB WORKER DOSE AND ANNUAL RWP HOURS IN RB

Legend

- Millirem per Hour
- Annual RWP Hours



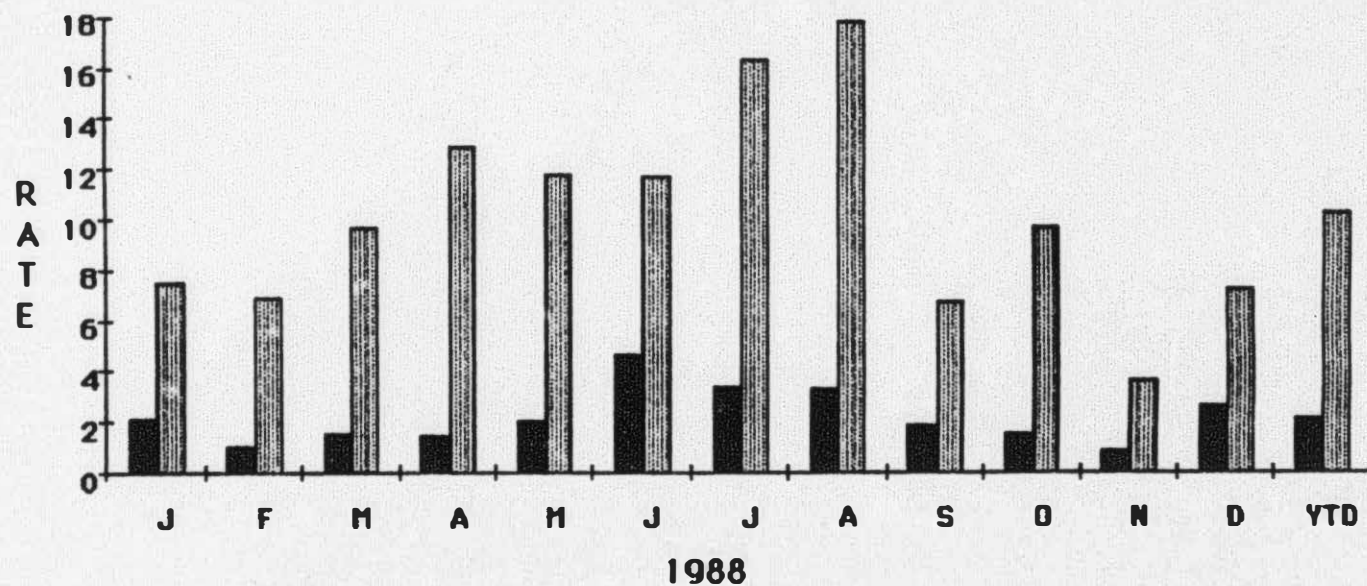
1989 TMI-2 PERSON-REM ESTIMATE

DEFUELING OPERATIONS (IN-VESSEL)	115
DEFUELING SUPPORT	185
BALANCE-OF-PLANT DECON	30
BALANCE-OF-PLANT OPERATIONS	50
MISC. ACTIVITIES	200
EX-VESSEL DEFUELING	<u>35</u>
TOTAL (IN PERSON-REM)	615

NOTE: ALL VALUES WERE DERIVED FROM CORRECTED TLD VALUES

SKIN CONTAMINATION RATE PER 10,000 RWP HOURS WORKED

■ >10 MREM
▨ <10 MREM



TMI-2 CLEANUP PROGRAM

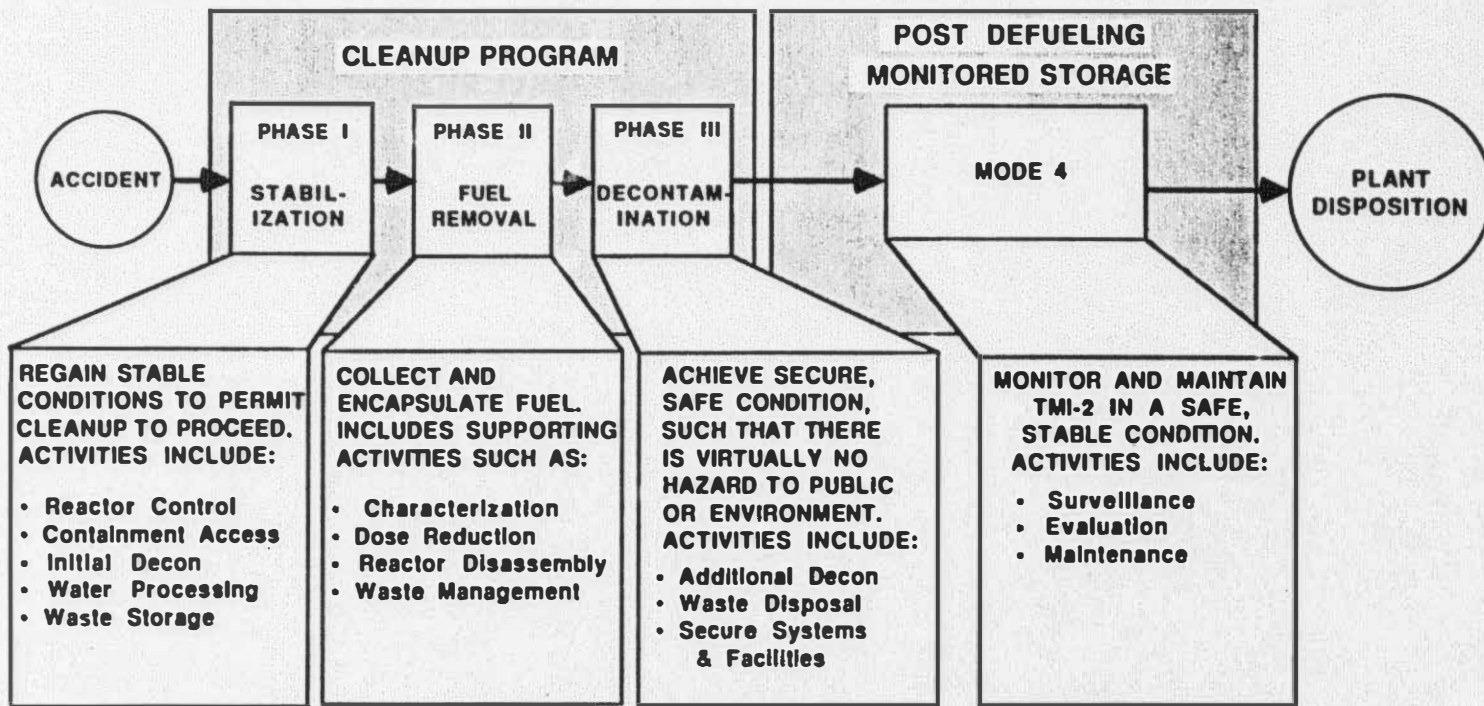
POST - DEFUELING

MONITORED STORAGE

(MODE 4)

POST-DEFUELING STORAGE (MODE 4)

December 2, 1986	GPU Nuclear Submitted A Plan To NRC For Information
March 11, 1987	GPU Nuclear Submitted An Environmental Evaluation
April 27, 1988	NRC Draft PEIS Issued For Comment
August 16, 1988	Safety Analysis Report Submitted To NRC For Approval



Overall Cleanup Program Strategy

BASIS FOR TRANSITION TO FACILITY MODE 4

- THE REACTOR VESSEL AND THE REACTOR COOLANT SYSTEM HAVE BEEN DEFUELED AND THE CORE MATERIAL HAS BEEN SHIPPED OFFSITE.
- DECONTAMINATION HAS BEEN COMPLETED TO THE EXTENT THAT FURTHER MAJOR DECONTAMINATION PROGRAMS ARE NOT JUSTIFIED ON THE BASIS OF WORKER DOSE.
- A CONDITION OF STABILITY AND SAFETY HAS BEEN ESTABLISHED SUCH THAT THERE IS NO RISK TO PUBLIC HEALTH AND SAFETY.

CONCEPTUAL APPROACH TO **PDMS (MODE 4)**

- NOT A COMMITMENT TO DECOMMISSIONING
- MODE 4 CONDITIONS SHALL PROVIDE A MARGIN OF SAFETY WHICH EXCEEDS EXISTING REGULATIONS FOR OFFSITE EXPOSURE (10 CFR 50, APPENDIX I)
- MODE 4 CONFIGURATION SHALL BE ACCEPTABLE FOR EXTENDED PERIOD, E.G., UP TO TIME OF TMI-1 DECOMMISSIONING
- MODE 4 CONFIGURATION SHALL BE ESSENTIALLY A PASSIVE, MINIMUM MAINTENANCE CONFIGURATION MAXIMIZING USE OF EXISTING SYSTEMS

MODE 4 PROVIDES LAYERS OF PROTECTION

- INHERENT STABILITY
- EFFECTIVE CONTAINMENT
- POSITIVE CONTROL

INHERENT STABILITY

- THE PLANT WILL BE STABLE AND NOT OPEN TO TRANSIENTS.
- OVER 99% OF THE FUEL WILL HAVE BEEN REMOVED. A NUCLEAR CHAIN REACTION WILL NOT BE POSSIBLE.
- CONTAMINATION WILL HAVE BEEN REMOVED TO THE EXTENT THERE CAN BE NO HAZARDOUS RELEASE OF RADIOACTIVITY.
- WATER REMOVED FROM PLANT SYSTEMS AND EQUIPMENT.
- FEW COMBUSTIBLES, LOW FIRE POTENTIAL.
- NO PRESSURIZED SYSTEMS.

EFFECTIVE CONTAINMENT

REMAINING RADIOACTIVITY ISOLATED FROM THE ENVIRONMENT BY PROTECTIVE STRUCTURES:

- CLOSED PIPING SYSTEMS
- SEALED CUBICLES
- LOCKED REACTOR CONTAINMENT BUILDING
- SECURE AUXILIARY AND FUEL HANDLING BUILDINGS

POSITIVE CONTROL

- RADIOLOGICAL AND ENVIRONMENTAL MONITORING
- PLANT PROTECTION SYSTEMS, SUCH AS FIRE PROTECTION
- PLANT SECURITY

PDMS (MODE 4) LICENSE CONDITION

- CONTINUE LICENSE UNDER 10 CFR PART 50 - "POSSESSION ONLY"
- NEW NRC-APPROVED TECHNICAL SPECIFICATIONS FOR FACILITY MODE 4
- COMPLY WITH CURRENT RADIATION PROTECTION REGULATIONS - 10 CFR PART 20
- OFFSITE DOSE A SMALL FRACTION OF 10 CFR PART 50, APPENDIX I LIMITS

PREREQUISITES FOR FACILITY MODE 4

- THERE IS NO CREDIBLE POSSIBILITY OF NUCLEAR CRITICALITY. THIS CONDITION HAS BEEN ASSURED BY REMOVAL OF SUBSTANTIALLY ALL OF THE FUEL AND ELIMINATION OF ALL POTENTIALLY CRITICAL CONFIGURATIONS.
- ALL FUEL AND CORE DEBRIS WHICH HAVE BEEN REMOVED FROM THE REACTOR VESSEL AND ASSOCIATED SYSTEMS HAS BEEN SHIPPED OFFSITE.
- ANY POTENTIAL FOR A SIGNIFICANT RELEASE OF RADIOACTIVE MATERIAL HAS BEEN ELIMINATED. RADIOACTIVE MATERIAL HAS BEEN REMOVED AND RESIDUAL SOURCES OF RADIOACTIVITY HAVE BEEN ISOLATED SO THAT ANY POTENTIAL RADIOACTIVE RELEASE WILL BE WITHIN 10 CFR 50 APPENDIX I GUIDELINES FOR OFFSITE DOSE CONSEQUENCES.

PREREQUISITES FOR FACILITY

MODE 4

(CONTINUED)

- TO THE EXTENT PRACTICAL, THE REACTOR COOLANT SYSTEM AND THE FUEL TRANSFER CANAL HAVE BEEN DRAINED, AND THE SPENT FUEL TRANSFER TUBES HAVE BEEN ISOLATED. TO THE EXTENT THAT THE SPENT FUEL POOLS ARE NEEDED TO SUPPORT ACCIDENT GENERATED WATER DISPOSAL ACTIVITIES, WATER MAY REMAIN IN THESE POOLS SUBSEQUENT TO TRANSITION TO MODE 4.
- ALL RADIOACTIVE WASTE FROM THE MAJOR CLEANUP ACTIVITIES HAS BEEN SHIPPED OFFSITE OR HAS BEEN PACKAGED AND STAGED FOR SHIPMENT OFFSITE.
- RADIATION WITHIN THE FACILITY HAS BEEN REDUCED, CONSISTENT WITH ALARA PRINCIPLES, TO LEVELS WHICH WILL ALLOW NECESSARY PLANT MONITORING ACTIVITIES, THE PERFORMANCE OF REQUIRED MAINTENANCE, AND ANY NECESSARY INSPECTIONS.

NUCLEAR CORE DEBRIS PROJECTIONS

DAMAGED CORE PLUS STRUCTURE = 296,900 LB

- DAMAGED CORE 134,050 KG
- REMOVE AND SHIP > 132,700 KG
- RESIDUAL FUEL (UO₂) < 940 KG
- RESIDUAL CORE DEBRIS < 1,340 KG

TMI-2 CLEANUP PROGRAM **RADIOLOGICAL GOALS**

GENERAL AREA
DOSE RATE
R/HR

REACTOR BUILDING

Refueling Canal	< 0.015
El. 347' & Above	< 0.03
(except D-rings)	
El. 305' to 347'	< 0.07
Basement (El. 282')	As Is

AUXILIARY BUILDING

Corridors	< 0.0025
Other Areas	< 0.05

TMI-2 CLEANUP PROGRAM
RADIOLOGICAL GOALS
(CONTINUED)

GENERAL AREA
DOSE RATE
R/HR

FUEL HANDLING BUILDING

Corridors	< 0.0025
Other Areas	< 0.05

OTHER BUILDINGS

Turbine Building	< 0.0025
Chemical Cleaning Bldg. (except EPICOR II area to be left operable)	< 0.0025
Service Building Containment Tank Area	< 0.0025

ESTIMATED DOSE COMMITMENTS
ROUTINE AND UNANTICIPATED
CONDITIONS DURING MODE 4

1. POPULATION DOSES

PERSON-REM/YEAR

Routine Airborne Releases:

Bone Dose	0.27
Total Body Dose	0.07

Routine Liquid Releases:

Bone Dose	0.09
Total Body Dose	0.02

Total Population Dose:

Bone Dose	0.36
Total Body Dose	0.09

ESTIMATED DOSE COMMITMENTS
ROUTINE AND UNANTICIPATED
CONDITIONS DURING MODE 4
(CONTINUED)

2. <u>DOSES TO MAXIMALLY EXPOSED</u> <u>INDIVIDUAL</u>	<u>MREM/YEAR</u>
---	------------------

Routine Airborne Releases:

Bone Dose	0.06
Total Body Dose	0.02

Routine Liquid Releases:

Bone Dose	0.005
Total Body Dose	0.002

3. MAXIMUM UNANTICIPATED RELEASE

Dose to maximally exposed individual
from unanticipated release (bone dose
limiting)

4.8 mrem

RADIOLOGICAL AND ENVIRONMENTAL MONITORING

- SURVEYS CONDUCTED ON A REGULAR BASIS
- MONITOR BOTH LIQUID AND GASEOUS EFFLUENTS
- MAINTAIN ENVIRONMENTAL MONITORING PROGRAM (REMP)
- MONITORING AND ENVIRONMENTAL REPORTS TO THE NRC

TMI-2 FACILITIES & SYSTEMS

- **Original**
- **Recovery**

All Placed in One of Two Classifications

OPERABLE

These include:

Mode 4 Support

Systems and Facilities kept operational full time or available to operate to support Mode 4 activities

Site Support Systems

Systems and Facilities made available to support site operations

DEACTIVATED

These include:

Mothballed for Future Use

Systems and Facilities for which steps have been taken to preserve availability for future use

Not Preserved

Systems and Facilities safe, but with no action taken to preserve future availability; some systems may be partially or wholly dismantled

CLASSIFICATION OF FACILITIES AND SYSTEMS

FACILITIES AND SYSTEMS **(FACILITY MODE 4)**

- TMI-2 BOUNDARY
 - Included within TMI protected area
 - External security equivalent to operating plant

- REACTOR BUILDING
 - Normally locked but accessible
 - Containment maintained
 - Ventilation system operable
 - Sump systems operable

- AUXILIARY AND FUEL HANDLING BUILDINGS
 - Normally locked but accessible
 - Exhaust systems operable
 - Sump systems operable

MODE 4 ACTIVITIES

- MONITORING AND SURVEILLANCE
- DECONTAMINATION
- RADIOACTIVE WASTE HANDLING
- SNM ACCOUNTABILITY
- WATER PROCESSING
 - AGW
 - Other

ENVIRONMENTAL EVALUATION FOR TMI-2 POST-DEFUELING MONITORED STORAGE (FACILITY MODE 4)

CONCLUSIONS

1. POTENTIAL OFFSITE CONSEQUENCES DURING MODE 4 DEMONSTRATE THREAT TO PUBLIC HEALTH AND SAFETY HAS BEEN ELIMINATED.
2. OFFSITE CONSEQUENCE ESTIMATES FOR MODE 4 CONFIRM PEIS CONCLUSION THAT POTENTIAL ENVIRONMENTAL CONSEQUENCES ARE DOMINATED BY OCCUPATIONAL EXPOSURES.
3. OCCUPATIONAL EXPOSURES DURING CLEANUP EXPECTED TO BE SUBSTANTIALLY LOWER THAN REVISED PEIS ESTIMATES; MAY BE WITHIN RANGE OF ORIGINAL ESTIMATES.
4. MODE 4 ACTIVITIES RESULT IN INSIGNIFICANT INCREASE IN TOTAL OCCUPATIONAL EXPOSURE.
5. OCCUPATIONAL EXPOSURE DURING RECOMMISSIONING/DECOMMISSIONING WILL BE REDUCED DUE TO NATURAL DECAY AND IMPROVED TECHNOLOGY.

ABSTRACT

DRAFT PEIS SUPPLEMENT 3

THE NRC STAFF HAS CONCLUDED, BASED ON THIS EVALUATION, THAT THE LICENSEE'S PROPOSED PLAN AND THE NRC STAFF IDENTIFIED ALTERNATIVES FOR COMPLETION OF CLEANUP ARE WITHIN THE APPLICABLE REGULATORY LIMITS AND COULD EACH BE IMPLEMENTED WITHOUT A SIGNIFICANT ENVIRONMENT IMPACT. NO ALTERNATIVE WAS FOUND TO BE CLEARLY PREFERABLE FROM AN ENVIRONMENTAL IMPACT PERSPECTIVE.

ABSTRACT

DRAFT PEIS SUPPLEMENT 3

THE NRC STAFF HAS CONDUCTED THAT THE LICENSEE'S PROPOSAL TO PLACE THE FACILITY IN A MONITORED STORAGE CONFIGURATION WILL NOT SIGNIFICANTLY AFFECT THE QUALITY OF THE HUMAN ENVIRONMENT. FURTHER, ANY IMPACTS FROM THE LONG-TERM STORAGE OF THE FACILITY ARE OUTWEIGHED BY ITS BENEFITS.

POST-DEFUELING STORAGE **(MODE 4)**

SAFETY ANALYSIS REPORT

SUBMITTED AUGUST 16, 1988

SUBMITTAL INCLUDES:

- Proposed License Change "To Possess" Only
- Proposed Technical Specifications For Facility Mode 4
- Supporting Safety Analysis Report

OTHER LICENSING BASIS DOCUMENTS

- Security Plan, Revision 26
- Emergency Plan, Revision 2
- QA Plan
- Organization Plan

SUMMARY

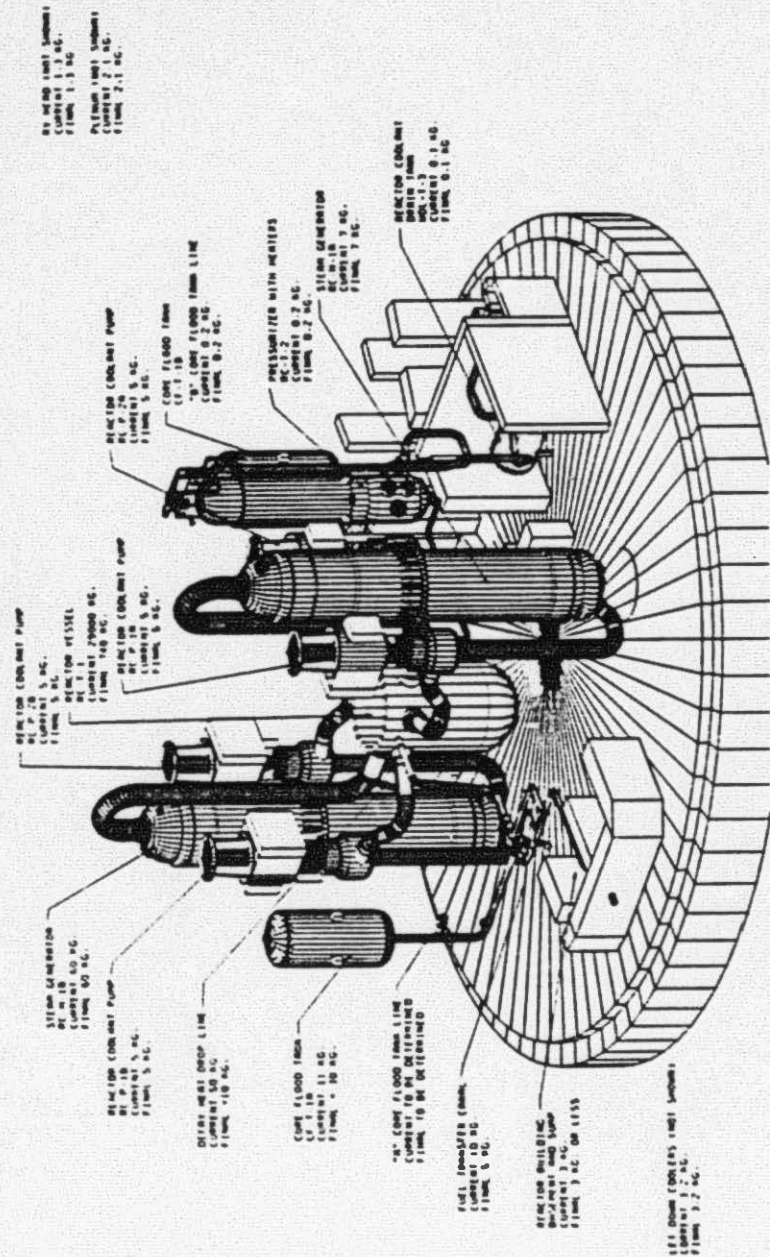
THE FACILITY MODE 4 (PDMS) CONFIGURATION
PROVIDES A SECURE, MONITORED TMI-2 PLANT
THAT DOES NOT PRESENT A HAZARD TO
WORKERS OR TO PUBLIC HEALTH AND SAFETY.

ISSUES

- RESIDUAL FUEL
- HIGH LEVELS OF RESIDUAL
CONTAMINATION
- FUTURE PLANT DISPOSITION

RESIDUAL FUEL INVENTORY
TMI-2

010 - 0 Ball: 1/6/89



PRX. / FUEL HANDL. BLOC.	CURRENT	FINAL COAL
	*30 MC	*20 MC.

REACTOR BUILDING		
	CURRENT INVENTORY	FINAL GOAL
IN VESSEL	19,000 KG.	6740 KG.
EX VESSEL	1,000 KG.	180 KG.

Diagram illustrating the Sellafield gas-graphite gas-cooled reactor (GGR) system, showing the reactor vessel, primary and secondary loops, steam generator, condenser, and various tanks and lines.

REACTOR VESSEL
RC-1-1
CURRENT 35,000 GAL.
FINAL +50 GAL.

STEAM GENERATOR
RC-N-10
PRIMARY SIDE
CURRENT 15,000 GAL.
FINAL 0 GAL.
SECONDARY SIDE
CURRENT 14,000 GAL.
FINAL +50 GAL.

CONDENSER
RC-1-2
CURRENT 2300 GAL.
FINAL 0 GAL.

STEAM GENERATOR
RC-N-11
PRIMARY SIDE
CURRENT 15,000 GAL.
FINAL 0 GAL.
SECONDARY SIDE
CURRENT 6000 GAL.
FINAL +50 GAL.

REACTOR COOLANT
DRAIN TANK
MDL-1-3
CURRENT 200 GAL.
FINAL 200 GAL.

PRESSURIZER SURGE LINE

DECAY HEAT DROPLINE

REACTOR BUILDING
SUMP
CURRENT 1000 GAL.
FINAL 2070 GAL.

CORE FLOOD TANK
CF-1-1A
CURRENT 8000 GAL.
FINAL 0 GAL.

CORE FLOOD TANK
CF-1-1B
CURRENT 0 GAL.
FINAL 0 GAL.

STEAM GENERATOR J-LEG

WATER VOLUMES TO BE REMOVED FROM LOGS OF RCS

	CURRENT	PLAN
PRESSURIZER SURGE LINE	10 CAL.	0 CAL.
COND FLOOD TRAP LINES 12	500 CAL.	1 TOTAL
DECAT HEAT DROP LINE	200 CAL.	0 CAL.
870 LOWER HEAD 12	120 CAL.	1 TOTAL
370 J-LEGS 141	600 CAL.	1 TOTAL

[illegible]

ATTACHMENT 2

TMI-2 CLEAN-UP STATUS MEETING

JANUARY 11, 1989

ATTENDANCE LIST

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