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January 11, 1991

U. S. Nuclear Regulatory Commission
Document Control Desk
1 White Flint North
11555 Rockville Pike
Rockville, MD 20852

Gentlemen:

SUBJECT: Cintichem Response to NRC Request for Additional Information Regarding Decommissioning Plan - Cintichem, Inc.

This correspondence details our response to the above referenced NRC letter dated December 21, 1990.

Item 1

Provide, for each structure noted in items 23-1 through 33-4 of Table 1.2a, the volume of structural contamination and the curie content by isotope. Provide similar information for contaminated systems and components such as the primary reactor cooling system, the primary reactor cooling purification system, the reactor building air exhaust system, the exterior air discharge duct and the stack, and the waste water evaporator system. Provide the volume of waste associated with each component and structural element in Table 1.4 such as the reactor core support tower, the reactor grid plate and locator pins, the plenum, the core outlet assembly, the pneumatic rabbit assembly, the beam tubes, and the thermal column and the thermal column lead shield assembly.

Cintichem Response

An estimate of the radioactive waste volume and curie content for items 23-1 through 33-4 of Table 1.2(a) resulting from decontamination of structural areas and from dismantling of contaminated systems is given in Table 1a enclosed. Table 1b presents a waste volume estimate of the items listed in Table 1.4 of the Decommissioning Plan.

50-54
70-687

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140128

ADD: T. Michaels NF12

Table 1b

Activated Waste Volume from Table 1.4 Decommissioning Plan

Item	Material	In-situ Volume (C.F.)
Fuel Support Plate	Al 6061	1.5
Fuel Support Plate Alignment Pins	304SS	0.01
Core Support Tower	Al 6061	1.9
Plenum	Al 6061	6.5
Flapper Valve	Al 1100	0.11
Bellows	304SS	1.86
Thermal Column Lead Shield	Pb	1.9
Thermal Column Lead Shield Liner	Al 6061	0.2
Thermal Column External Liner	Al 1100	4.3
Pool Lead Liner	Pb	1.5
Beam Tubes	Al 1100	122.3
Pool Walls	Barytes	425.7
Pool Floor	Std. Conc.	186.0
Reinforcing Steel	Carbon Steel	1.5
	Total volume, metal and concrete	755.28

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Item 2

It is unclear from Section 1.3.5 where contamination was found and where it was not found. Provide such information. In addition, provide a layout indicating where subsurface contamination was found. Include with the layout an estimate of the volume of contaminated soil and the curie content of these volumes by isotope.

Cintichem Response

The location of structurally contaminated surfaces (floors, walls, structural steel, etc.) is shown on enclosed Figure 2a.

The locations of known subsurface contamination are shown on enclosed Figure 2b. These locations account for an estimated in-situ, undisturbed volume of 28,500 cubic feet of soil. Additionally, the radioactive waste volume estimate in the Decommissioning Plan includes 15,300 cubic feet of in-situ, undisturbed soil that is suspected of being potentially contaminated. This suspected soil is located such that it cannot be feasibly sampled until decommissioning activities expose these areas. The suspected areas include, under the primary reactor storage tank, under the 5K storage tanks and adjacent to various buried yard piping, as shown on Figure 2c.

The estimated in-situ volumes and radioactivity of soil from known subsurface contaminated areas is shown on the following page.

Item 3

The Decommissioning Plan does not indicate whether the quality assurance guidelines of Regulatory Guide 1.33, with respect to decommissioning, will be met. Will these guidelines be met?

Cintichem Response

Cintichem will not be using Regulatory Guide 1.33 as the basis for our Quality Assurance (QA) Program, since it applies to an operating facility. We will establish and implement our own written QA program designed specifically for the Cintichem decommissioning project.

The Quality Assurance Program, as applied to activities shall comply with and be responsive to applicable regulatory requirements and applicable industry codes and standards. These activities are for the protection of the health and safety of the public and project personnel, and for adherence to regulations and commitments made to the Nuclear Regulatory Commission, including the control of personnel exposure to radiation, control of radioactive material and contamination, and radwaste shipment.

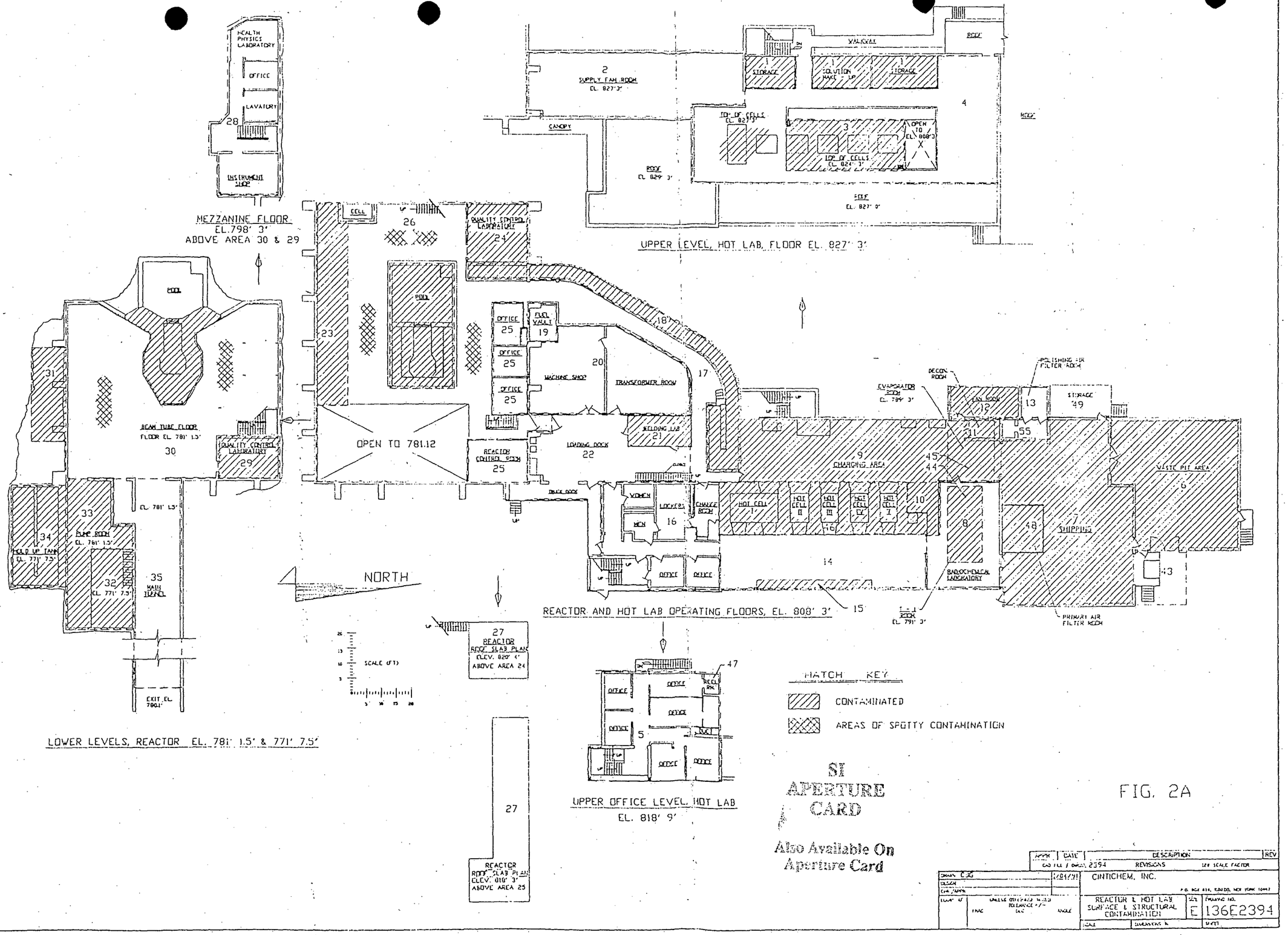


FIG. 2A

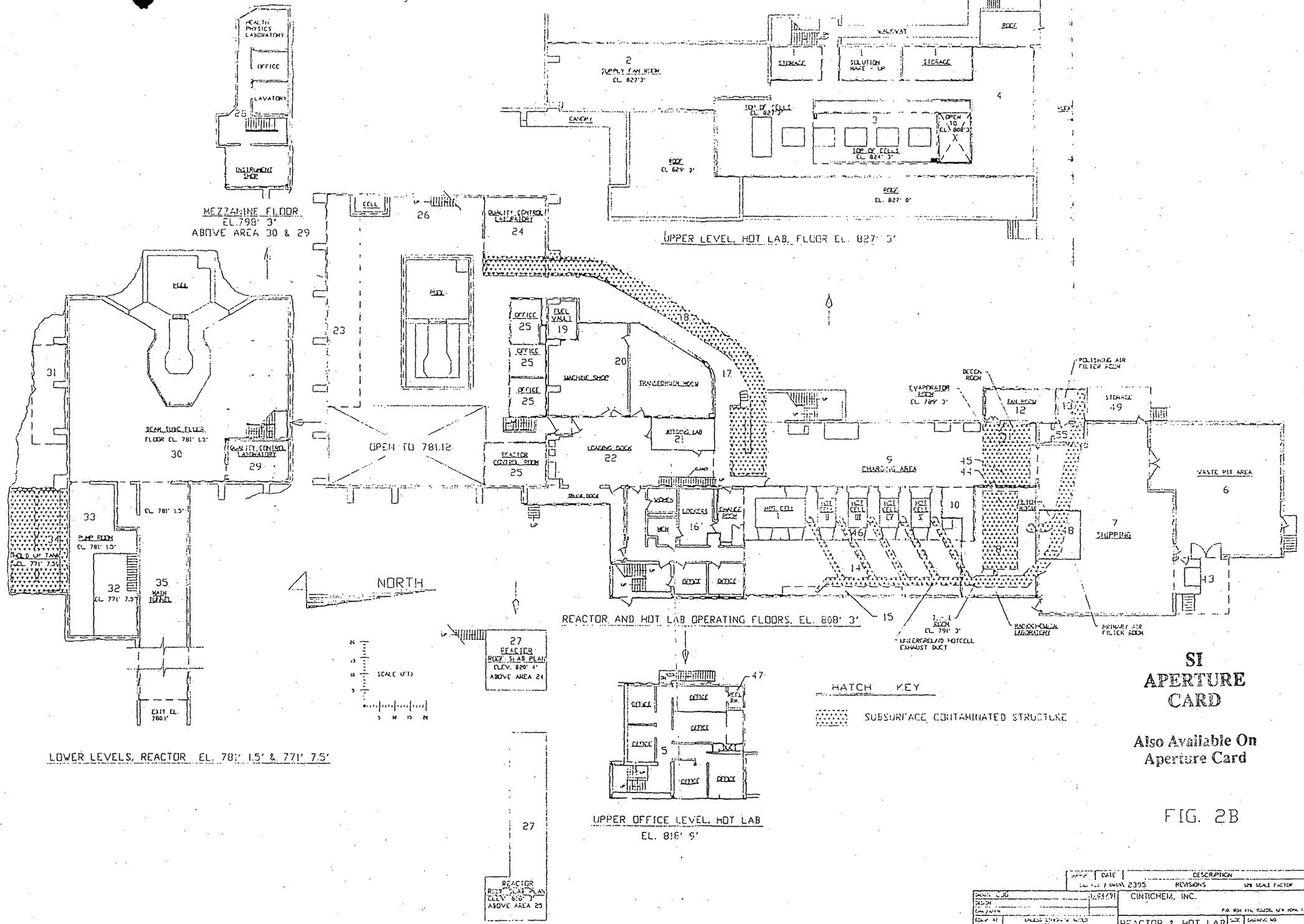
MARK	DATE	DESCRIPTION	REV
		REVISIONS	

CINTICHEM, INC.

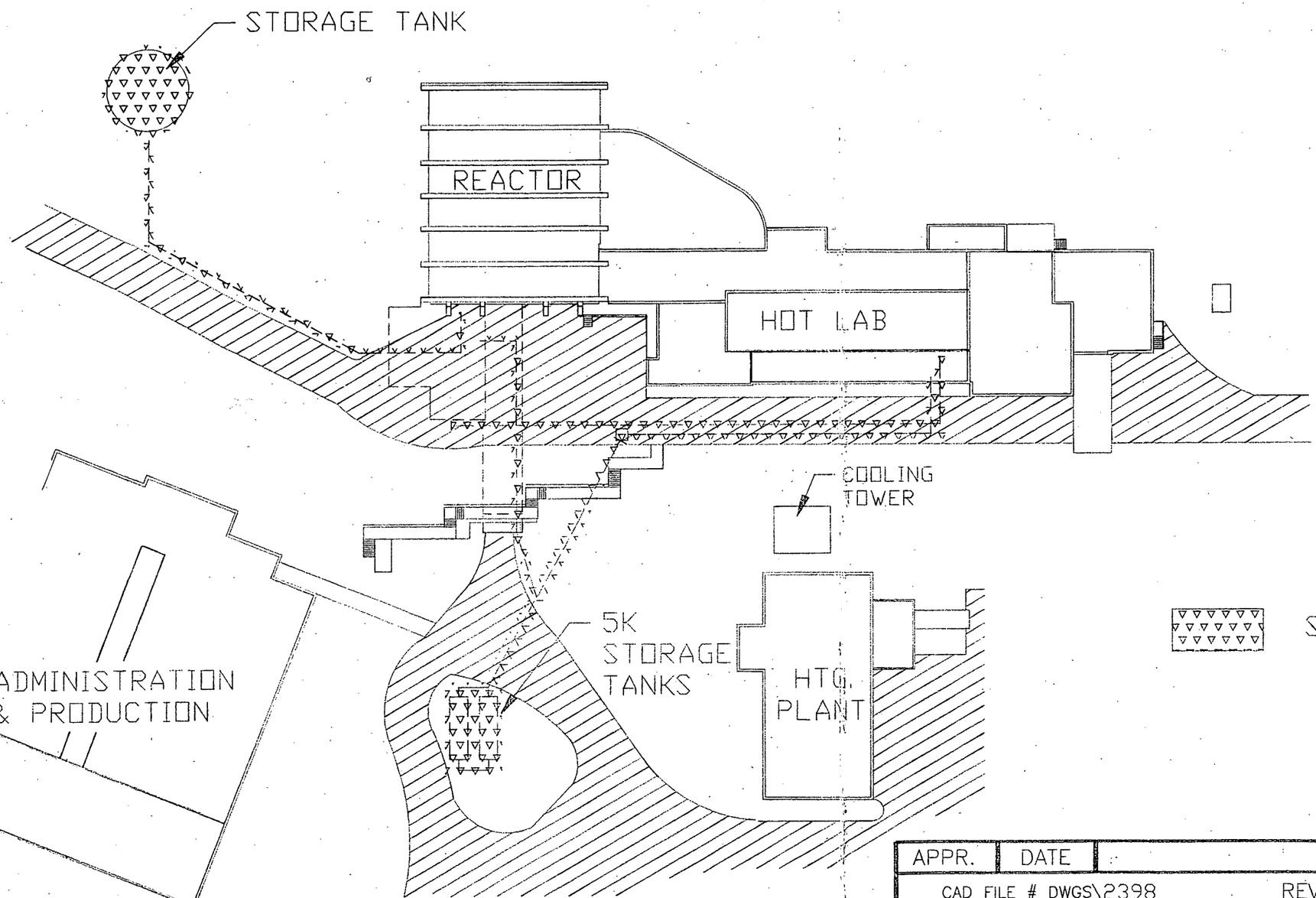
REACTOR & HOT LAB SURFACE & STRUCTURAL CONTAMINATION

DRAWING NO. 136E2394

9101150078-01



9101150078-02



SUSPECTED CONTAMINATION

FIG. 2C

APPR.	DATE	DESCRIPTION	REV
CAD FILE # DWGS\2398		REVISIONS	

DRAWN CJG	1/11/91	CINTICHEM, INC.
DESIGN		P.O. BOX 816, TUXEDO, NEW YORK 10987
CHK./APPR.		

EQUIP. AT

UNLESS OTHERWISE NOTED: TOLERANCE +/−
FRAC DEC ANGLE

SUSPECTED SOIL CONTAMINATION AREA	SIZE B	DRAWING NO. 136B2398
SCALE 1" = 50'	DIMENSIONS IN	SHEET

9101150078-03

The estimated in-situ volumes and radioactivity of soil from known subsurface contaminated areas is as follows:

Location	Estimated in-situ Volume (CF)	Estimated Total Activity (Ci)(a)	Estimated Isotopic Breakdown(a) (Curies)	
1. Holdup Tank Area	1,700	8.27E-3	Sc-46	3.3E-4
			Co-60	1.0E-3
			Zn-65	6.6E-4
			Nb-95	1.8E-3
			Ag-110m	8.3E-4
			Ce-144	2.8E-3
			Eu-152	1.7E-4
			Ir-192	2.5E-4
			Mn-54, Fe-59, Cs-137, Ag108, Sn-113, Sb-124, Sb-125, Cs-134,	< 4.3E-4
2. Canal/gamma pit area	9,100	1.76E-2	Ce-144	9.5E-3
			K-40	2.6E-3
			Co-60	1.2E-3
			Zn-65	9.3E-4
			Nb-95	5.6E-4
			Mn-54	3.3E-4
			Zr-95	1.8E-4
			Sc-46	1.6E-4
			Ag-108	6.8E-4
			Ag-110m	7.5E-4
			Cs-137	1.7E-4
			Ir-92	2.0E-4
			Sb-125	9.0E-5
			Ru-103	9.5E-5
			Sn-113, Eu-152, Sb-124, Cs-134,	1.5E-4

(a) Radioactivity as of July 1990. Radionuclide inventory will be less at time of removal and disposal due to radioactive decay of short-lived radioisotopes (i.e., Sc-46, Ir-912, Sn-113, Sb-124).

Location	Estimated in-situ Volume (CF)	Estimated Total Activity (Ci)(a)	Estimated Isotopic Breakdown(a)	(Curies)
3. Underground Exhaust duct/ T-1 Room Area	17,700	5.87E0	Ce-144	5.5E-1
			Cs-137	4.2E0
			Cs-134	1.5E-1
			Co-60	1.0E-1
			Sr-90	8.4E-1

(a) Radioactivity as of July 1990. Radionuclide inventory will be less at time of removal and disposal due to radioactive decay of short-lived radioisotopes (i.e., Sc-46, Ir-912, Sn-113, Sb-124).

Project procedures shall provide for compliance with appropriate regulatory, statutory, license, and industry requirements. Specific quality assurance requirements and organizational responsibilities for implementation of these requirements shall be specified.

Compliance with this program and provisions of project procedures is mandatory for personnel with respect to Cintichem decommissioning activities which may affect quality or the health and safety of project personnel or the general public. Personnel shall, therefore, be familiar with the requirements and responsibilities of the program that are applicable to their individual activities and interfaces.

Item 4

The Cintichem facility presently has a radiological environmental program. Will the same program continue during the decommissioning and dismantling operations or will the program be modified to address these operations?

Cintichem Response

The Radiological Environmental Program described in the Plan is a modification of the existing program. Most notably, two additional particulate air samplers will be located in "a southerly and westerly direction from buildings 1 and 2 within the site perimeter" (p. 7.7). The five listed sampling stations are in existence now with collection and assay frequencies ranging from weekly to monthly. The Plan calls for all seven stations to have weekly collection and assay frequencies.

Item 5

Explain why the total curies for each component in Table 1.4 is not in agreement with the total curies presented in Table 1.3.

Cintichem Response

Table 1.3 in the Decommissioning Plan does not provide total curies for each component and therefore, there is nothing in Table 1.3 that should match Table 1.4. The specific activities given in Table 1.3 are in units of curies per gram. If the specific activity for each item in Table 1.3 is multiplied by its density and its given volume, this will equal the items radioactivity given in Table 1.4.

Item 6

Clarification should be provided in Section 3.3 as to the use of HEPA ventilation contamination control. When it is indicated that such control is being utilized, does that mean all the actions associated with a given task utilize HEPA ventilation contamination control or do only some of the actions utilize such control?

Cintichem Response

HEPA ventilation will be used whenever and wherever there will be a potential for generation of airborne radioactivity. This will generally include all activities that involve abrasive cutting, thermal cutting or scarification activities on contaminated or potentially contaminated systems or structures. Therefore, those actions within a task that meet the above criteria will require the use of HEPA ventilation for control of airborne radioactivity. Where use of containment tents and HEPA ventilation is not feasible, alternate containment techniques, such as glove bag containment will be used.

Item 7

Why aren't the activities associated with the removal of the thermal column liner incorporating the HEPA ventilation contamination control system, when such an activity incorporates the cutting of the steel panels and the cutting of the liner, which would presumably generate airborne radioactivity? A similar question is raised on the use of such a ventilation system for the tasks associated with the decontamination of the reactor pump room and the pool/stall and the removal of the storage tank, the reactor building exhaust ventilation system, and the canal/gamma pit.

Cintichem Response

HEPA ventilation **will** be used during the potentially airborne generating activities associated with thermal column liner removal, reactor pump room and pool/stall decontamination, and removal of the storage tank, canal-gamma pit and reactor building ventilation system removal. As was previously stated in the answer to Question 6, any activity that will or could generate airborne radioactivity will be performed with HEPA ventilation controls.

Item 8

What is the volume of liquid radwaste which will be generated during the decontamination and dismantling process, the isotopic concentration levels prior to and following processing and the total curies released offsite? What will the resultant doses be to the maximum exposed individual?

Cintichem Response

It is estimated that 506,000 gallons of liquid radioactive waste will be generated during decommissioning. This volume of liquid waste will come from three primary sources; hot cell washdown (6,000 gallons), draining of the reactor pool and canal (200,000 gallons), and from miscellaneous floor drains, showers, and sinks (300,000 gallons). Water from the hot cell washdown process will be evaporated and then solidified. This will result with a solid waste volume with the following curie content:

Isotope	Curies*
Cs-134	0.019
Cs-137	0.202
Ce-144	4.016
Nb-95	3.62
Zr-95	1.78
Sb-125	0.12

* As of July 16, 1990

Water resulting from the draining of the reactor pool and canal will be treated by filtration and deionization of the water. It is estimated that the pre- and post-treatment curie content will be as follows for the 200,000 gallons.

Isotope	Pre-Treatment (Ci)	Post-Treatment (Ci)
Co-60	1.26 E-4	1.26 E-5
Nb-95	5.0 E-5	5.0 E-6
Sb-125	5.6 E-5	5.6 E-6
Cs-137	7.1 E-5	7.1 E-6
Ce-144	2.52 E-4	2.52 E-5
K-40	1.62 E-4	1.62 E-5

Water from floor drains, showers, sinks and other miscellaneous sources will be treated by evaporation. It is estimated that the pre- and post-treatment curie contents will be as follows for the 300,000 gallons.

Isotope	Pre-Treatment (Ci)	Post-Treatment (Ci)
Co-60	1.28 E-6	4.3 E-9
Nb-95	2.27 E-6	3.0 E-9
Zr-95	2.44 E-6	2.5 E-8
Sb-125	4.69 E-7	6.6 E-9
Cs-134	1.33 E-6	1.5 E-9
Cs-137	9.44 E-5	5.8 E-9
Ce-144	2.38 E-5	2.1 E-8

The mixture calculates to 1.8×10^{-3} MPC. Assuming no dilution before drinking, and that the maximally exposed individual's entire water consumption is at the end of the pipe, the dose is 1.5×10^{-6} Rem/year.

$$1.8 \times 10^{-3} \text{ MPC} \times \frac{0.5 \text{ Rem}}{\text{MPC/year}} = 9.0 \times 10^{-4} \text{ Rem/year}$$

This dose is 1.8×10^{-3}% of the 0.5 Rem/year dose permitted offsite individuals. Therefore, the estimate of total dose for the entire 2.5 year decommissioning period is 2.3×10^{-3} Rem.

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Item 9

What type of process equipment will be utilized in the mobile radwaste system and describe its intended mode of operation?

Cintichem Response

The mobile radwaste system is designed to process low-level aqueous radwaste by filtration and ion-exchange. The mode of operation involves pumping radwaste through particulate filters and ion-exchange column(s). The effluent will be collected in a tank, checked for radioactivity and recycled if radioactivity is found that exceeds free release requirements.

Item 10

Provide the quantity, in curies, and the concentration, for each radionuclide, of gaseous releases.

Cintichem Response

The inventory of gaseous radioisotopes during decommissioning is zero Curies. There are also no gaseous isotopes having a solid form precursor which are on hot cell internal surfaces, etc. Therefore, the gaseous release for decommissioning is zero Curies. Per telephone conversation with John Hayes January 10, 1991, we understand that the NRC interprets gaseous to mean all airborne releases. Particulate releases are presented in the answer to Question 19.

Item 11

Provide the manner in which the various forms of solid wastes will be treated and prepared for shipment offsite. Present sections in the decommissioning plan do not discuss treatment of all wastes. For example, will evaporator sludges be solidified? If so, how will they be?

Cintichem Response

As indicated in the Decommissioning Plan, eight solid waste streams will result from the decommissioning operations. They are:

- o Contaminated soil and concrete debris (concrete dust and small chips)
- o Dry active waste (plastics, paper, filters, etc.)
- o Activated components
- o Contaminated piping and equipment
- o Activated concrete

- o Activated lead
- o Contaminated lead
- o Evaporator concentrates

These waste streams will be treated and prepared for shipment as follows:

CONTAMINATED SOIL AND CONCRETE DEBRIS

This material will be packaged directly into steel 55 gallon drums or LSA boxes. This material may be packed along with other material (such as concrete rubble or concrete pipe sections) to fill interstitial void spaces and maximize packaging efficiency. Also, soil and concrete debris will be compacted into shipping containers using standard construction industry soil tampers, where feasible.

DRY ACTIVE WASTE

Dry active waste will be packaged in steel 55 gallon drums or LSA boxes. This material may be sent to an off-site waste processor for volume reduction or will be added to waste containers of high density Class A waste material that may become weight limited prior to complete filling.

ACTIVATED COMPONENTS

These components will be packaged in high integrity containers (if Class B or C) or in steel 55 gallon drums or boxes (if Class A). These containers will be shipped in shielded casks as required. These components will be cut into manageable sized pieces to permit efficient packaging into these containers.

CONTAMINATED PIPING AND COMPONENTS

These items will be cut into manageable sized sections and packaged into re-usable sea-van cargo containers and/or LSA boxes for transport to a waste processor/recycling center.

ACTIVATED CONCRETE

Activated concrete will be packaged as a rubble. It will be packaged in steel drums and boxes or high integrity containers (as determined by 10 CFR 61 requirements). To maximize packaging efficiency, contaminated soil and/or concrete dust will be used as fill between the rubble voids.

ACTIVATED LEAD

Activated lead will be packaged separately from other wastes. This material would be packaged in a steel DOT 17-H 55 gallon drum or appropriate for use in the Cintichem shipping cask. This waste will be retained on-site or stored off-site by a licensed storage facility until a suitable mixed waste disposal facility becomes available.

CONTAMINATED LEAD

Contaminated lead will be directly packaged in steel 55 gallon drums or LSA boxes. This material will be sent to an off-site waste processing/recycle facility for decontamination.

EVAPORATOR CONCENTRATES

Evaporator concentrates will be solidified in steel 55 gallon drums using solidification media acceptable to the disposal site. The same techniques and processes as Cintichem currently uses will be followed.

Item 12

Section 7.3.1 states that the stack monitor will be used to measure the total radioactive airborne concentration leaving the reactor building and the hot laboratory until the ventilation system is disabled. The section then states that airborne radioactive effluent will be monitored as per Table 7.1 yet it is unclear from the Table how such monitoring will be implemented. Provide information on the replacement monitoring and include it in Table 7.1.

Cintichem Response

When the stack and associated monitoring equipment is disabled, no significant airborne environmental releases are expected from buildings 1 and 2. To monitor the environmental air at this time, the seven particulate air samplers discussed in answer #4 will be utilized. In addition, local air monitoring will be performed.

Item 13

Since a great deal of dust and particulates will be generated during the decommissioning operations, why doesn't the HEPA filter include a differential pressure gauge with an alarm instead of a weekly check to ensure that the filter is not clogged?

Cintichem Response

Both the reactor building and hot laboratory HEPA filter units will include differential pressure gauges and high delta-P alarms.

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Item 14

Has it been confirmed from the latest land use census that the garden pathway or other ingestion pathways such as the milk ingestion pathway are nonexistent?

Cintichem Response

Yes it has. According to the latest land use census, no gardens larger than 500 ft^2 of leafy vegetables exist within a five mile radius of the site. For the milk ingestion pathway, two dairy farms exist 4.5 miles from the site in the least predominant wind direction. Semi-annual milk samples have shown no detectable levels of radioactive iodine.

Item 15

Does the determination of total surface contamination levels of systems and equipment, as noted in Section 8.2.2.3, include both fixed and loose contamination?

Cintichem Response

Yes, the determination of total surface contamination levels of systems and equipment includes both fixed and loose contamination. The analytical methods are described in Section 8.2.2.3 on page 8.8 of the Plan.

Item 16

Why were the accident doses calculated at the nearest residential development instead of at the nearest site boundary? Provide these same analyses for the nearest site boundary.

Cintichem Response

A higher dose was obtained by considering an individual located at the nearest residential development compared to the dose expected to someone located at the nearest site boundary. This is the reason the residential development dose was included in the Decommissioning Plan.

The following additional accident analysis for a teenager at the site boundary follows the format of Appendix E entitled "Estimation of Radiation Dose Due to Accidental Cutting of Activated Reactor Components". The calculated exposure is for a teen at the nearest site boundary located in a southeasterly direction at 850 feet from the reactor exhaust stack. The dose to a teen at the residential development (see response to question 17) is 0.0008 Rem. This dose is reduced by the following ratio at the site boundary location described above.

Teen Dose (Site Boundary) =

$$\text{Teen Dose (Residential Site)} \times \frac{\chi/Q \text{ (Nearest Site Boundary)}}{\chi/Q \text{ (Residential Site)}}$$

which is,

Teen Dose (Site Boundary) =

$$\text{Teen Dose (Residential Site)} \times \frac{(1/\pi \sigma_y \sigma_z u)\, e^{-\frac{1}{2}\left(\frac{H}{\sigma_z}\right)^2}}{1/\pi \sigma'_y \sigma'_z u} \quad (1)$$

The terms are:

π = 3.14
σ_y = 60 m (Nearest Site Boundary at 259 m) Class A see (2)
σ_z = 44 m (Nearest Site Boundary at 259 m) Class A see (2)
u = 1 m/sec (Average Wind Speed)
H = 60 m (Height of Plume Centerline above Nearest Site Boundary)
σ'_y = 18 m (Residential Development at 457 m) Class F see (2)
σ'_z = 7.8 m (Residential Development at 457 m) Class F see (2)

Class F wind type is used for the Residential Development component of the calculation whereas Class A is considered at the Nearest Site Boundary. These assumptions yield the highest doses at the respective locations.

The calculation works out as:

Teen Dose (Site Boundary) = 0.0008 Rem x 0.02 = 0.00002 Rem

Which demonstrates that the higher dose would be expected to occur at the Residential Development.

(1) Turner, D.B. - Workbook of Atmospheric Dispersion Estimates. U. S. Department of Health, Education and Welfare. Public Health Service Publication No. 999-AP-26. Revised 1969. p. 6 (equation 3.3).

(2) Slade, D.H. ed., Meteorology and Atomic Energy, July 1968, pp 102 - 103.

Item 17

Why did one accident analysis involving airborne releases have the dose calculated to a teenager while the other analysis calculate the dose to a child? Provide the dose to the maximum exposed individual.

Cintichem Response

Four accidents are analyzed in the decommissioning "plan". The "plan" and "actual" critical receptors are listed for each accident.

Accident	Plan Critical Receptor	Actual Critical Receptor
1. Accidental cutting of activated reactor components (Case 1 child/residential area)	Child (adult breathing rate)	Teen
2. Accidental cutting of activated reactor components (Case 2 Adult/reactor bldg.)	Adult	Adult
3. Resuspension of hot cell concrete dust (Case 1 worker/outside)	Adult	Adult
4. Resuspension of hot cell concrete dust (Case 2 Teenager/site boundary)	Teen	Teen

For accident 1 above, the child was considered the critical receptor in the plan. A conservative overestimate of the dose to the child was made for the plan calculation by assigning an adult breathing rate. The result is an effective dose equivalent of 0.0014 Rem. Although this approach may be considered unconventional, it did result in the most conservative dose. We should have considered the teenager to be the critical receptor and the calculation would be as follows:

Teen Dose

$$\text{Co-60 dose} = (4.28 \times 10^{6}\text{pCi/m}^3)(3.3 \times 10^{-4}\text{m}^3\text{/sec})(428\text{ sec})(1.09 \times 10^{-3}\text{ mRem/pCi})(0.01) = 6.59\text{ mRem}$$

$$\text{Zn-65 dose} = (2.04 \times 10^{5}\text{pCi/m}^3)(3.3 \times 10^{-4}\text{m}^3\text{/sec})(428\text{ sec})(2.69 \times 10^{-4}\text{ mRem/pCi})(0.01) = 0.31\text{ mRem}$$

for a total of 6.90 mRem = 0.0069 Rem

Since the "effective dose equivalent" weighting factor is 0.12 for the lung, the estimated equivalent total body dose is:

$$0.12 \times 0.0069\text{ Rem} = 0.0008\text{ Rem}$$

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This value is less than 0.2 percent of the applicable limit of 0.5 Rem. Note also that this more exacting method yields a dose which is 57% of the dose reported in the plan (i.e. 0.0008/0.0014 x 100 = 57%).

Item 18

It was indicated in Section 5 that an accident analysis was conducted of flooding after the reactor building and the hot laboratory were breached. However, the analysis presented in Appendix G only evaluated flooding of the hot laboratory and then only the contamination associated with the sediment was considered thereby, giving the impression that the accident considered is actually following the razing of the building. Was an analysis performed of the impact of flooding of the above two buildings immediately after they had been breached? In other words, the buildings are still intact except for the breach point. Was contamination associated with the structures, equipment, components, and systems remaining in the buildings following the breaching of the buildings included in the evaluation and found not to be the worst case? Provide the results of such analyses. Why was the evaluation to demonstrate compliance with Appendix B of Part 20, calculated with the concentration of the release based upon dilution of the Indian Kill Reservoir and not calculated based upon the concentration of the water at the restricted area boundary in accordance with 10 CFR 20.106(d).

Cintichem Response

The accident presented in Section 5.0 is the accident detailed in Appendix G. This accident considered flooding of the exposed excavation that would result from razing of the buildings. Flooding accidents are not considered possible prior to razing the buildings, when structural and system contamination are still present.

Decontamination of structures and contaminated soil removal will proceed with the structures intact. The buildings would only be razed with remaining soil contamination if prior removal would undermine the structural integrity of the buildings. Razing of the building would not be performed without NRC concurrence. Therefore, the worst case accident scenario involved contamination in soil and on footings that might have to remain until structures have been removed.

Small breeches may be needed during D & D work, such as removal of a duct penetration or enlarging a door opening. If so, the breech would be covered with a structurally sound containment barrier to preclude in-flow of water and maintain the building's air containment.

The postulated accidental release of water would entail an instantaneous release of a large quantity of water. This water would rush down the hill towards the reservoir. It is not credible that any of this water could be consumed until it collected in the reservoir.

Item 19

The information presented in the decommissioning plan on the dose to the maximum exposed individual should not have been based upon the performance of a single task but rather on the performance of all tasks. A calculation should have been performed of the effluents associated with all of the decommissioning and dismantling tasks involving the potential discharge of radioactive materials. This information should have been provided in the licensee's decommissioning plan along with the related doses. Therefore, provide the total estimated effluents from decommissioning and their related doses. The doses should be calculated at the site boundary with the highest X/Q value and at the actual receptor location which combines the most sensitive ingestion pathway and highest deposition rate, if applicable.

Cintichem Response

The total estimated effluent is provided below. Refer to Table 1A attached entitled "Structural and System Waste Volume and Isotopic Curie Content". The Curie total for decommissioning work other than hot cell scarification is 1.00×10^{-2} Ci. The total on hot cell surfaces is 6.80 Ci consisting of equal amounts of Sr-90 and Cs-137 (see Attachment A in Decommissioning Plan).

An assumption is made that the potential for airborne release for Table 1A work is the same as that for hot cell scarification. This is a conservative assumption since not all of the work in Table 1A involves scarification although the majority does. Since all scarification work air effluent is filtered by two HEPA filters in series (each rated at a minimum of 99% efficiency), the processes are additive in terms of offsite dose potential. Although the mixture of isotopes in Table 1A has a lower dose factor than a 50/50 Sr-90 and Cs-137 mix, the latter dose factor is assumed.

The conclusion is that effective source term for all work on the site is increased by the ratio:

$$\frac{\text{Hot Cell Ci + Table 1A Ci}}{\text{Hot Cell Ci}}$$

The numerical value is:

$$\frac{6.80 \text{ Ci} + 0.01 \text{ Ci}}{6.80 \text{ Ci}} = 1.001$$

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All analyses for offsite dose have been made to two significant figures. Therefore, no change is necessary in what has been already reported in the Decommissioning Plan.

Please refer to our response to question #16 which shows that the most limiting offsite location is the Residential Development rather than the Nearest Site Boundary. Also note that the most current land use census precludes soil deposition from concern and therefore inhalation is effectively the total pathway.

Item 20

On page 2.12 it states that "All significant dismantling operations will be controlled by written procedures." What defines an operation as significant?

Cintichem Response

A written work procedure will be required for work which could result in spread of radioactive contamination, airborne radioactivity or increased radiation levels, or which, if improperly performed, could result in a radiological incident. A radiological incident being the following:

- external radiation exposure in excess of administrative limits
- internally deposited radioactivity in excess of administrative limits
- skin contamination
- personnel exposure to excessive airborne radioactivity without respiratory protection
- unauthorized discharge of radioactivity to the environment
- lost or uncontrolled radioactive material
- spread of radioactive contamination outside of radiologically controlled areas
- improper control of radiation or contamination control areas

A work procedure is therefore required for work which involves direct contact with actually or potentially contaminated or activated items or areas. Observations, inspections or similar operations do not require written procedures if there is no direct contact.

Item 21

In numerous places there are statements saying that a certain material (e.g. "inner building wall" on p. 3.20 and "oil between windows" on p. 3.21) will be disposed as clean material. Will these materials (and others already identified as clean material) be verified as being "clean" before release?

Cintichem Response

All materials that are free released will be verified as "meeting release criteria at the time of release".

Item 22

Has soil where water was collected during accidental release in 1989 and 1990 been surveyed? If not, should this be added as an area for a biased environmental survey?

Cintichem Response

This soil has been assayed. It was not included in the decommissioning site survey because it is covered by the remediation investigation conducted by the NYSDEC.

Item 23

Section 1.0 and 1.4 says the New York state license will be terminated, which is inconsistent with Section 1.1.1 which says that operations will be discontinued only in Buildings 1, 2 and 6. Building 4 is under the N.Y. state license. Please clarify.

Cintichem Response

The New York state license will be amended to omit operations in buildings 1, 2 and 6 once decommissioning has been completed. The operations in building 4 are expected to continue after the scheduled decommissioning.

Item 24

Discuss the target fretting and leaking problem identified in the 1983 - 1984 time period.

Cintichem Response

During October and November of 1983, there were two uranium target failures during irradiation that resulted in transient contamination of the reactor building and the reactor primary coolant system with short-lived fission-product radionuclides. These incidents can be reviewed in the attached reports dated November 30, 1983 and December 30, 1983.

JJM/09.91B

The consequences of these target failures included airborne contamination of Xenon-138 and Kr-88. Very little iodine became airborne due to its solubility in the primary coolant. Tests for uranium in the primary coolant revealed nothing greater than our LLD by delayed neutron analysis (approximately 1 x 10^{-11} gms/gallon or 2 x 10^{-5} gms total dissolved uranium). It is believed that some uranium oxide (UO_2) was released, but it remained a solid and got swept into the hold-up tank. There was no appreciable increase in the building airborne radioactivity during subsequent routine operation of the reactor further indicating the absence of uranium contamination in the primary coolant. Any insoluble uranium that originated from these target failures would have been removed from the hold-up tank or the pool system during subsequent routine cleaning.

These contamination incidents were deemed to be unrelated to the decommissioning project because there should be no appreciable residual contamination that would have to be removed over and above that which has already been identified in the plan.

Item 25

In Section 1.3.1, shouldn't the outside Primary Water Holdup tank be listed as contaminated and discussed in Section 1.3.2?

Cintichem Response

The primary water holdup tank is considered part of the primary reactor cooling system discussed on page 1.14 of the Plan and listed under Contaminated Systems and Equipment on page 1.11 of the Plan.

Item 26

Shouldn't Section 1.3.2 also address Building 6?

Cintichem Response

Section 1.3.2 is a summary description of contaminated structures. Building 6 is expected to have only minimal contamination, if any, since it only stores waste containers ready for shipment. Fixed contamination characterization is impossible at this time due to the areas elevated exposure rate from the materials awaiting shipment. Once all waste has been removed from this building a complete and thorough contamination characterization will be performed and the area will be decontaminated before it is free released.

Item 27

Section 1.6 should also include NUREG CR-5512, "Residual Radioactive Contamination from Decommissioning" in the informal guidance.

JJM/09.91B

Cintichem Response

Section 1.6 which lists regulations, regulatory guides and standards will now include NUREG CR-5512.

We trust that the information provided herein will be satisfactory responses to the questions posed in your letter referenced above. Following your review and acceptance we will modify the Decommissioning Plan to incorporate these modifications and additions.

Very truly yours,

J. J. McGovern
President/Plant Manager

JJMcG/bjc

Attachments

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JJM/09.91B

UNION CARBIDE

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To (Name) F. J. Morse
Division Secretary, NSC
Location
Area

Date November 29, 1983

Originating Dept. NUCLEAR OPERATIONS
Author W. G. Ruzicka
Area

Copy to C. J. Konnerth - Chairman
K. D. George

D. J. Gallaher
D. D. Grogan
S. E. Lupinski
J. J. McGovern
R. H. Quackenbush
R. A. Strack
L. C. Thelin
Reactor Console

Subject Fission Product Molybdenum (FPM) Target Capsule Rupture

Abstract:

A second FPM capsule ruptured in the core. See W. G. Ruzicka's 10/26/83 report to the Nuclear Safeguards Committee for details of the first rupture. This capsule had been in the core only 4 minutes prior to the rupture. The rupture of the capsule initiated the excursion monitor and the building evacuation sequence. No personnel involved with the incident were contaminated or received a dose exceeding 45 mRem whole body. The total release of airborne radioactivity to the environment was less than 1 percent of the Technical Specification limits.

Sequence of Events:

Wednesday 11/16/83	2333	FPM capsule No. -831 is loaded into the pool. Reactor Operations had prior to the pool loading satisfactorily bubble leak checked capsule No. -831.
	2336	FPM capsule No. -831 is loaded into core position E3 No. 4. The reactor remained at 5 MW during the capsule loading.
	2340	The N16 high level alarmed, several building 1 radiation area monitors alarmed, and the reactor building evacuation sequence is automatically initiated by the bridge 5R/hr. monitor. The reactor is manually scrammed via the console scram button and the reactor and hot laboratories are evacuated. Offsite Nuclear Operations and Health Physics personnel are informed of the incident.

Thursday
11/17/83 0041 Nuclear Operations and Health Physics personnel enter the hot laboratory for initial surveys. The stack monitor shows no increase in gaseous, particulate, or iodine. The hot lab CAMS read approximately 42,000 cpm. There is no unusual direct radiation readings in the hot lab. Direct radiation readings around the reactor building are 50-60 mR/hr. at the inner airlock door. The particulate filter in the emergency reactor building exhaust system reads approximately 100 mR/hr. on contact.

0138 An air sample is taken of the reactor building through a building 2 access port.

0330 The NRC is informed of the incident.

0350 Nuclear Operations and Health Physics enter the reactor building to take a direct air sample.

0430 Health Physics determines that the building 1 airborne radiation levels are due mainly to Xenon and Krypton gases. These decay to the particulate daughters Cesium-138 and Rubidium-88 having an average half life of 22 minutes. This indicates fission product noble gas was released into the building.

0440 Reactor Operations enters the reactor building and resets the ventilation system.

0745 Unrestricted access to the reactor building is permitted. The Health Physics requirement for protective clothing and masks is discontinued.

0850 All FPM capsules in stringer E3 are checked. All are not bound in the stringer except the target in E3 No. 4.

0854 FPM capsules from E3 No. 1, E3 No. 2, and E3 No. 3 are unloaded. The E3 single pull stringer and E3 No. 4 FPM capsule No. -831 are removed from the core and encapsulated in a sealed container.

0930 The container holding the ruptured capsule and its stringer is taken to the building 2 gamma pit. Capsule No. -831 is removed from its stringer and taken into hot cell No. 1. Capsule No. -831 can be seen now to have a axial line rupture approximately 2" long and approximately in the center of the capsule.

Cont'd
Thursday
11/17/83 1000 A Nuclear Safeguards Committee meeting is held to review the second target failure incident. The Committee recommends that the reactor be restarted without FPM targets in the core and the uranium contamination in the pool water be determined via delayed neutron analysis. A review of the single pull stringer heat transfer calculations is also to be performed.

1600 A second Nuclear Safeguards Committee meeting is held. The review of the heat transfer calculations for the single pull stringer identified no problems with the initial calculations. Although no problems were identified the Committee recommends that if the delayed neutron count is satisfactory the reactor should be restarted using the old box design FPM stringers. The Committee also agreed that the reactor should not be restarted with FPM capsules installed until another Safeguards Committee meeting is held on Friday morning.

2000 The reactor is started up. All FPM targets have been removed from the core and all single pull stringers are loaded with dummy capsules to restrict flow through the stringers.

2130 Delayed neutron counting indicates no abnormal pool water uranium levels.

2145 The reactor is shut down.

Friday
11/18/83 0100 All single pull stringer and dummy FPM capsules are removed from the core.

0400 FPM capsules are reloaded into the core in old box type design stringers.

0845 A third Nuclear Safeguards Committee meeting is held. The results of the delayed neutron count showing no pool water uranium is reviewed. The Committee recommends that the reactor can be restarted with FPM capsules in the old stringers. The percent fit calculations for the FPM capsules will be redone utilizing the box stringer flow parameters. An extra 10 percent reduction in percent fit will also be required on all capsules until further studies have been concluded. The reactor is to be brought to 100 percent power in steps of 10 percent for an hour, then 50 percent for an hour, then to full power.

Friday Night/
Saturday Morning

The reactor is brought slowly up to full power with FPM targets in the old box stringers. No problems with FPM tagets is noted.

Safety Analysis Report Result Comparisons:

The NRC issued a Safety Evaluation Report for UCNR Single Encapsulation for In-Core Iodine Production and Iodine Increase to 1000 Curies Per Capsule. This report concluded that, following an in-core capsule failure, only 1/10000 of the iodine released would reach the surface. If a 1000 Curies I-131 equivalent capsule ruptured the report predicted that .1 Curie of Iodine would reach the surface of the reactor pool water.

The consequences of this second target failure can be compared to the predicted consequences of the Safety Analysis Report. Target No. -831 had been irradiated for 4 minutes. The significant radioactive iodine inventory that was contained in this target at the time of rupture was I-131 .05 Curies, I-133 1.24 Curies.

The measurements of these radionuclides which were released to the air in the building are as follows:

I-131	$< 3.4 \times 10^{-11}$ μCi/cc	(1/265 MPC)
I-133	$< 8.0 \times 10^{-10}$ μCi/cc	(1/38 MPC)

Capsule No. -831 had approximately .1 Curies of iodine inventory (I-131 equivalent). Health Physics measured 0.5 μ Curies of iodine (I-131 equivalent) to have been released into the building containment. The ratio of capsule iodine inventory to iodine in the building air was 200,000 to 1. This was less than 1/20 of the level predicted in the NRC Safety Evaluation Report.

Conclusion from Radioactive Material Release Measurements:

- No person involved in the incident received a whole body dose exceeding 45 mrem. Most of the exposure was caused by noble gas activity.
- Radioactive material releases to the environment were less than 1 percent of the Technical Specification limits.
- Average building 1 air activity for the whole event was less than the permissible concentration for 40 hour occupancy (excluding dose from noble gas cloud).
- The total iodine (I-131 equivalent) released was less than 0.001 Curies.

Analysis of Target Failure:

The second F.P. Mo-99 production target failed 11/16. It failed in a manner that suggests a cause other than internal pressure. The failed target exhibited a break in the wall about 2" long axially at about the middle of the target. This break is centered within a patch of color that is distinct from the remainder of the surface color. It had been irradiated for only 4 minutes before failure. This second target failure and the frequent incidence of similar discoloration on other targets indicates that some overheating of these targets may have caused these failures.

Although the single pull stringer has been in use for slightly more than a year, it was decided to revert to the former box stringer design as an interim corrective action and to reduce the maximum allowable heat flux to ~ 31 watts/cm^2 (90 percent of previous limit) until the heat transfer effectiveness of this system can be verified.

Overheating can be caused by two basic mechanisms; namely:

a. Higher than anticipated fission rate within the target.
b. Less than anticipated heat removal by core coolant.

Following the second failure, the initial investigative work involved the following:

a. Check of heat transfer calculations.
b. Estimate flow reduction or heat flux increase that would cause boiling.
c. Remeasure neutron fluxes
d. Recheck fission product yields.

K.D. George, using a variation of the Lane Correlation, confirmed the margin below boiling that exists with the limit of 37.5 W/cm^2 and the reduction in flow that would be necessary to cause nucleate boiling. A summary is as follows:

Basic Data:

Heat Flux = 37.5^2 W/cm

De = .15 in^2

A = .317 in^2 (radial gap = .076 in)

ΔT_B to centerline of target = 3.4°C at 3.7 fpsv

$T_B = 38 + \Delta T_B$ (°C)

$$\Delta T_f = 266/[V^{.8}(1 + \frac{T_B}{100} - \frac{T_B^2}{10^5})]\,^\circ C \quad (T_B \text{ here in } ^\circ F)$$

$T_s = T_B + \Delta T_f$ (°C)

T_p = 38°C Pool

Results are tabulated:

V(+T/sec)	ΔT_B(°C)	T_B(°C)	ΔT_f(°C)	T_s(°C)	
3.7	3.4	41.4	48	89	
2.5	5.0	43	65	108	
2.0	6.3	44.3	77	121	(T_s(boil)=123°)

Since the failed target was operating at 75 percent of the 37.5 W/cm^2 limit and the coolant velocity was ~ 3.2 fps it can be seen that this target did not exceed the design boiling surface temperature.

The above correlation can be checked by the basic Lane Correlation for the film heat transfer coefficient and also by another correlation developed by Stein and Begell.

Lane

$$h = .0964 \left[1 + \left(\frac{T_B}{100}\right) - 10^{-5} (T_B)^2\right] V^{.8}/De^{.2}$$

$$h = .0964 \left[1 + \frac{110}{100} - \frac{(110)^2}{10^5}\right] 3.5^{.8}/ .15^{.2}$$

$$h = .0964 [2.1 - .12] 2.72/.684$$

$h \cong .76$ watts/cm^2 °C $\cong$ 1340 BTU/hr °F ft^2

$$\therefore \quad \Delta T_f = \frac{37}{.76} = 48.7°C$$

V = ft/sec

$$De = \frac{4A\ (in)}{P}$$

t = °F

h = watts/cm^2 °C

Stein and Begell

$$h = 97.7 [1 + .0066 (t_f)] V^{.8}/De^{.2} \left(\frac{D_2}{D_1}\right)^{1/2}$$

$$h = 97.7 [1 + .0066 (150)] \left(\frac{3.5^{.8}}{.013^{.2}}\right)\left(\frac{.1167}{.1042}\right)^{1/2}$$

$$h = 97.7 (1 + .99) \left(\frac{2.72}{.420}\right)(1.058)$$

h = 1332 BTU/hr ft^2 °F

Heat Flux = 37.5 watts/cm^2 => 1.19 x 10^5 BTU/hr ft^2

$$\Delta T_f = \frac{1.19 \times 10^5}{1332} = 89°F\ (49.6°C)$$

$$T_f = °F = \frac{T_B + T_s}{2}$$

V = ft/sec

D_e = ft

D_2 = .1167 ft

D_1 = .1042 ft

h = BTU units

This is in very good agreement with the previous two correlations. The margin to boiling is confirmed.

These same correlations can be used to estimate what increase in heat flux is required to achieve boiling:

t_{sat} = 117°C (at core level of pool)

$T_{s(boil)} = T_{sat}$ + superheat

$= 117 + 6 = 123°C$

$\Delta t_{f(boil)} = T_{s(boil)} - t_b$

$= 123 - 41 = 82°C$

Heat Flux $_{(boil)} \cong (\Delta t_{f(boil)})(h)$

$\cong$ 82°C x .76 watts/cm^2°C $\cong$ 62 watt/cm^2 or 168 percent of limit

It is not conceivable that the fission rate could have exceeded the design limit by 168 percent.

A further check of the literature confirms that our heat flux ($\sim$ 1 x 10^5 BTU/hr ft^2 or 33 watts/cm^2) is well below the burnout heat flux. Empirical data from tests approximating our system is tabulated from Gambill's work:[1]

	MIT	USSR	UCC
ΔT sub ($T_{sat} - T_b$)	14-99	14-313	120
Pressure	30-90	28-278	29
Velocity	1-20	4.5-13.4	3.5
Burnout	.5-2x10^6	.6-6x10^6	
UCC			.1x10^6

It can be seen that our flux limit is 1/5 the lowest measured burnout flux under approximately the same conditions.

The failures observed do not indicate a true burnout condition; there is no evidence of melting. Given these margins it is apparent that some peculiarity may exist with the single-pull stringer that causes overheating. The appearance of the failure (axial splitting) suggests a hoop stress or tension on the outer wall surface. The thermal stress caused by the ΔT within the wall

1 Generalized Prediction of Burnout Heat Flux for Flowing, Subcooled, Wetting Liquids, TID7642, 112-137, 1963.

had previously been estimated at ~ 15,000 psi maximum. This stress is developed with a $\Delta T \cong 40°F$. The ΔT to achieve the yield stress for this alloy (30,000 psi) can be estimated:

$$\frac{E}{2(1-V)} \; a \; .96(\Delta T) = 30{,}000$$

E = modulus Elast. (2.9×10^7)

a = linear expansion coeff. (9.2×10^{-6})

V = Pousson's rates (.28)

$$\frac{267}{1.44} \times .96(\Delta T) = 30{,}000$$

$$176(\Delta T) = 30{,}000$$

$$\Delta T \cong 170°F$$

It is conceivable that rapid heating and cooling could cause such a ΔT and thereby cause failure from thermal stress.

It is also evident that this condition developed very recently because similar operating conditions of flow, target power, pool temperature and reactor power existed on many prior occasions without incident. If the coolant channel gap were to change, local boiling could start from reduced flow causing transient heat transfer conditions. The channel gap could change from movement of the target within the stringer or warping of the target during irradiation. Such a change would reduce flow locally. Irradiated targets were gauged and found to be as true as they were prior to irradiation, thereby eliminating any permanent warping to be the cause. Wear of the positioning lugs in the stringer could cause the target to be off-center in the channel.

Some literature [1] indicates that buoyant forces in narrow channels ($L/De > 200$) at velocities < 8 fps could cause reduced flow. Our $L/De = 106$ but our velocity is 3.5 fps. This should be pursued.

Some evidence exists that surface texture can effect h. The latest batches of tubing that were used for target production have had a very smooth outer surface.

Planned Further Investigations:

Although the cause of the target failures has not been positively identified, the remedial actions taken so far have shown some positive results in that the post-irradiation discoloration is markedly reduced. Continuing investigation is planned as follows:

a. Irradiate an instrumented target (thermo-couple monitor on surface) to verify the heat transfer correlation being used.
b. Check heated tube for warping.
c. Gauge used single pull stringers for wear of bosses.
d. Investigate the influence of surface texture on heat transfer (wetting influence).
e. Continue investigating buoyancy phenomenon.
f. Irradiate targets with very clean surfaces.
g. Reevaluate K factor (process yields).

APPENDIX A

RECALCULATION OF FIT LIMIT FOR TRANSITION (BOX) STRINGER WITH SINGLY ENCAPSULATED TARGETS

The current bulk coolant flow rate through the core is 2300 $\leq$ 2200 GPM. The ΔP for a 2200 GPM flow rate can be estimated:

$$\frac{.7}{\Delta P \text{ for 2200 GPM flow}} = \frac{2400^2}{2200^2}$$

$$\frac{.7}{\Delta P} = 1.19$$

$$\Delta P = \frac{.7}{.19} = .59$$

From flow curves .59 DP = > 36 GPM (no orifice)

36 GPM x .002228 = .08 ft^3/sec.

$$\frac{0.08 \text{ ft}^3/\text{sec}}{.0222 \text{ ft}^2} = 3.6 \text{ ft/sec}$$

Basic Data:

t (sat) = 117°C
Superheat = 6°C (for incipient boiling)
ts (boil) = 123°C
Max pool temp = 130°F = 54.4°C

$$h = .188 \frac{V^{.8}}{De^{.2}} \quad (De = .815 \text{ in}) \quad (h = \text{watt/cm}^2\text{°C})$$

$$h = .188 \frac{(3.6)^{.8}}{(.815)^{.2}} = \frac{.188\ (2.786)}{.96} = .546 \text{ watts/cm}^2\text{°C}$$

.86 .612

ΔTB = 6°C (bulk coolant rise)

For non boiling $\Delta Tf + \Delta TB + T \text{ pool} \leq 123°C$

$$\Delta Tf + 6 + 54.4 \leq 123°C$$

$$\Delta Tf \leq 62.6°C$$

$$\Delta Tf = \frac{\text{heat flux}}{h} = \frac{\text{heat flux}}{.546} \leq 62.6°C$$

Therefore the maximum heat flux allowed at 2200 gpm with no boiling is 34.2 watts/cm^2. (For a core rate of 2100 gpm, core ΔP of .52 and stringer flow of 33.5 gpm the maximum heat flux allowed with no boiling is 32.3 watts/cm^2.)

APPENDIX B

DERIVATION OF EQUATION APPLYING HEAT TRANSFER LIMIT TO FPM TARGETS

Given:

Heat flux limit at plating	= 34.2 w/cm^2 (2200 gpm core flow)
Primary target ID	= 1.18 in.
Plated length	= 16 in.
Fission cross section for U-235 at 38°C	= 568 barns
Heat per fission deposited in target	= 175 meV

Fission rate = $\phi = n\sigma\phi$

Heat generation rate = $n\sigma\phi \times 175$ meV

$$= \left[\frac{M(\text{g U-235})\,6.023\times10^{23}\ \frac{\text{(atoms)}}{\text{mole}}}{235\ \frac{\text{(gm U-235)}}{\text{mole}}}\right] \times \left[\phi\ (10^{13}\ \frac{\text{neutrons}}{\text{cm}^2\ \text{sec.}})\right]$$

$$\times \left[586 \times 10^{-24}(\text{cm}^2)\right] \times \left[175\ \text{meV} \times 1.586 \times 10^{13}\ \frac{\text{w-sec.}}{\text{meV}}\right]$$

$$= 404\ M\phi\ (\text{watt})$$

Heat transfer area = $\pi\ d\ h$

$$= \pi \times 1.18" \times 16" \times 2.54^2\ \frac{\text{cm}^2}{\text{in}^2}$$

$$= 383\ \text{cm}^2$$

$$\text{Heat flux} = \frac{404\ M\phi\ (\text{watts})}{383\ \text{cm}^2} = 1.06\ M\phi\ (\text{w/cm}^2)$$

Equation for maximum target loading:

Heat flux $\leq$ heat flux limit

$$1.06\ M\phi \leq 34.2\ (w/cm^2)$$

$$M\phi \leq 32.26$$

The above equation assumes the plating, M, is uniform and the flux at the plating, ϕ, is known. Incorporating the real situation, the same principal can be applied locally for peak conditions as follows:

$$M\left(\frac{C_p}{C_a}\right) \times \phi \text{ actual} = \frac{M\left(\frac{C_p}{C_a}\right)\phi_m}{\frac{\phi_m}{\phi}} = \frac{M C_p \phi_m}{C_a K} \leq 32.26$$

where: ϕm = measured flux

C_p = peak plating density

C_a = average plating density

$$K = \frac{\phi_m}{\phi}$$

The target is evaluated over five representative (but not necessarily equal) increments with the equation in the form:

$$\frac{M\frac{C_i}{C_a}\phi_{mi}}{K} \leq 32.26$$

The above equation assumes a 2200 gpm core flow. If 2100 gpm core flow is given then the equation is as shown below

$$\frac{M\frac{C_i}{C_a}\phi_{mi}}{K} \leq 30.5$$