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COMMUNITY SAFETY DEPARTMENT
OFFICE OF RESEARCH & OCCUPATIONAL SAFETY
LOS ANGELES, CALIFORNIA 90024

October 8, 1985

Mr. Harold Denton, Director
Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attn: SSPB

Docket No. 50-142
License No. R-71

Dear Mr. Denton:

In my last letter to you dated March 13, 1985, I indicated that UCLA was at that time developing plans for dismantling and decommissioning our reactor. I further indicated that we were moving toward having all of the work done by an outside contractor.

During the spring and summer months we have had discussions with, and received informal bids from a number of potential contractors. We have performed more detailed analyses of the radiological status of much of the remaining parts of the reactor (see Appendix A attached). Finally, we have formulated more detailed plans of what steps need to be done to dismantle the reactor structure itself (see Appendix B attached), preparatory to decommissioning the entire facility. As a result of this work we now plan to do the initial dismantling in-house, under my direction.

The necessary technical and engineering support will be provided by the Nuclear Energy Laboratory (NEL) staff and health physics support will be provided by the staff of the UCLA Radiation Safety Office (see Appendix C attached). The health physics staff is a part of my normal safety staff (Office of Research and Occupational Safety) and for the duration of the dismantling operations the NEL staff will report to me. The necessary labor will be provided by "Decontamination Technicians" hired through Allied Nuclear Corporation of Fremont, California. I want to point out that the disassembly (or dismantling) that needs to be done has been done a number of times in the past (including replacement of the entire fuel box assemblies) by the NEL staff. Further, the packaging and transport of low level radioactive materials (either for burial or transfer to another licensee) is a routine operation of UCLA's Radiation Safety Office. Therefore, we are confident that we have all of the expertise "in-house," that will be required to complete the dismantling of our reactor.

The dismantling will in general simply proceed as the reverse of assembly, i.e., graphite and lead will be unstacked and pipes and structures will be unbolted. We plan to minimize cutting of

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radioactive material and whatever cutting is done will be saw cuts (no torch burning) to minimize airborne contamination.

In the past, reactor disassembly and reassembly (including fuel removal and replacement) has been accomplished with a total radiation dose of 35-40 man-rem, after a three week cooling off time from reactor shutdown. The reactor has now been shutdown for 21 months and operated only briefly during the 12 months prior to final shutdown. Further, all fuel has been removed and of course, no reassembly is possible. Therefore, I estimate that the dismantling operations can be done with a total dose of perhaps 10% (3-4 man-rem) of that incurred in past disassembly/reassembly operations.

Finally, virtually no airborne radioactivity will be generated, minimal contaminated liquids will be generated (only from personnel decontamination), and the radiation sources present generate very low external radiation fields (see Appendix A), therefore there are no potential sources of radiation exposure to the public.

We want to start this dismantling work by November 4, 1985 (to avoid conflict with construction of a new engineering building). Therefore, we would greatly appreciate anything you can do to expedite NRC approval of our plans.

Sincerely,

Walter F. Wegst, Jr.
Director, Office of
Research & Occupational Safety

WFW:kf

Enclosure

cc: Director, U.S. NRC Region V

STATE OF CALIFORNIA)
COUNTY OF LOS ANGELES)

On October 9, 1985, before me, the undersigned, a Notary Public in and for said County and State, personally appeared WALTER F. WEGST, JR. known to me to be the person whose name is subscribed to the within instrument and acknowledged to me that he executed the same.

WITNESS my hand and official seal

Notary Public in and for
said County and State

OFFICIAL SEAL
R. WANDA SOUKUP
NOTARY PUBLIC CALIFORNIA
LOS ANGELES COUNTY
My Commission Expires June 10, 1988

APPENDIX A: PRELIMINARY RADIATION SURVEY

A.1 ACTIVATION PRODUCTS IN GRAPHITE AND LEAD

Samples of graphite and lead from the reactor core have been examined with a GeLi detector and multichannel analyzer to determine specific activities and radioisotopic composition.

Graphite stringers taken from the central region of the reactor core in March of 1985 exibited surface radiation levels from 20 to 50 mr/hr near the center of those vertical four-foot stringers. At a perpendicular distance of one foot from the center of the stringers, the level fell to 5 to 8 mr/hr. Samples of graphite taken from locations within the central region near core mid-height were found to contain 13.6 year Eu-152 as the principal radioisotope. Observed specific activities ranged from 0.3 to 0.45 micro-Ci per gram (3-20-85). Eu-154 and Co-60 were also observed, each at a specific activity about one order of magnitude less than that of the Eu-152.

The lead above the reactor core consists of two layers of lead bricks, each layer is two inches thick. Samples from the upper layer at a corner of the reactor were measurably radio-active (0.5 mr/hr on the surface), and silver-110 was identified at a concentration of about 240 pico-Ci per gram. Bricks taken from the lower course in the vicinity of the fuel boxes were appreciably more radioactive (5 mr/hr on the surface). The principal isotopes are silver-108 (127 yr) and silver-110 (252 day). The results for six samples, three from each of two bricks, are shown in table A-1.

Table A-1: Activities in Lead (5-13-85)

Sample		Ag-108	Ag-110
1-1	11.3 gms	110	2510
1-2	30.3 gms	186	3410
1-3	17.4 gms	158	3100
	Average	164	3100
2-1	38.0 gms	91	2300
2-2	20.5 gms	87	2600
2-3	20.3 gms	173	3600
	Average	110	2670

Activities are in pico-Ci per gram, the "averages" are the mass-weighted average for each brick.

A.2 ACTIVATION PRODUUCTS IN CONCRETE

The reactor beam port plugs are composed of concrete cast in aluminum sleeves. Samples taken from the south beam port plug and counted on May 15, 1985, showed the following activation products (in order of dominance):

Table A-2: Activation Products in Concrete (5-15-85)

	Isotope	Activity*	%
1.	Eu-152, 13.6 y	183	52
2.	Co-60, 5.3 y	157	42
3.	Eu-154, 8.6 y	19	5
4.	Mn-54, 312 d	9	2
5.	Cs-134, 2.1 y	9	2
	TOTAL	377	100

*Activities are in nano-curies per gram. The results in Table A-2 pertain to a sample taken one inch from the interior fac of the plug (four inches into the core from the biological shield).

The total activity, as a function of distance from the inner face of the biological shield was found to be:

Table A-3: Specific Activity, Concrete (5-15-85)

Distance, inches	Activity, n Ci/gm
-4	377
-2	229
0 (Note 1)	203
2	126
4	36.6
7	11.5
13	1.46 (Note 2)
19	Background

Note 1: Zero is taken as the inner face of the concrete biological shield.

Note 2: At the 13 inch depth, Eu-154 and Cs-134 were not discernable. The sum of the other isotopes identified in Table A-2 was divided by 0.93 to obtain the activation product concentration of 1.46 nCi/gm. In that same sample a number of short-lived radium-thorium daughter products appeared.

A least-squares best fit of an exponential to the five points from zero to nineteen inches indicates a relaxation length (e-folding distance) of about 2.2 inches for the activation products in this concrete.

The concrete was black in color, evidently the plugs are filled with magnetite concrete. It is premature to assume that the samples are representative of all of the concrete and higher levels of activity may appear due to streaming neutrons through the control blade shrouds or along pipeways. This question cannot be answered without unstacking the core.

A.3 RADIATION SURVEY OF THE SOUTH BEAM PORT

A radiation survey (by Teletector) of the south beam port in December of 1984 yielded the following readings as a function of distance from the internal end. The "zero" distance is 16 inches interior to the inner face of the biological shield; i.e. the 16 inch measurement is at the inner surface of the biological shield.

Table A-4: South Beam Port Observations

Distance, inches	Radiation, m rem/hr
0	3000
8	2000
16	650
28	55
52	4.5

The high readings correspond to locations within a few inches of the steel blade drive support bearings and structure.

A.4 RADIATION MEASUREMENT IN CORE CENTER VOID

The reactor core was uncovered to the top of the fuel boxes in March 1985 for a radiation survey and to collect samples. Fifteen vertical graphite stringers (4 inches by 4 inches by 4 feet) were removed thereby creating a void 12 inches by 20 inches in horizontal cross-section and four feet deep. The radiation field in the void was observed to have a nearly uniform value of one rem per hour.

A.5 ACTIVATION OF CORE METALLIC PARTS

Other than lead, aluminum is the predominant metallic core component. The fuel boxes, the shield tank, and various plumbing and tubing are composed of aluminum. The vertical port liners are composed of type 6061 aluminum which according to NUREG/CR-1756, vol 2, Table E.1-1, contains 0.25% zinc. A small sample of a vertical port liner examined with the GeLi detector showed both Zn-65 (244 day) and Co-60 (5.27 yr) in the atomic ratio of about 2:3. The calculated total activity based upon the reactor operating history followed by 20 months decay is about six micro-Curies per gram. The calculated result agrees well with a measured radiation field of 10 mR/hr at a distance of three inches from the tube center line at a location far from either end of the tube.

The highest concentrations of radioisotopes are expected in the structural steel and the small amount of stainless steel in the core. Calculated values for these activities, based upon the spatial-maximum, time-averaged neutron flux, are shown in Table A-5.

Table A-5: Calculated Activities in Steels

Structural Steel	Fe-55	1.2 mCi/gm
Stainless Steel	Fe-55	0.8 mCi/gm
	Co-60	0.3 mCi/gm
	Ni-63	0.1 mCi/gm

The majority of the steel parts are not in the highest flux region of the reactor, and the indicated values are high upper-bound estimates.

APPENDIX B: PRELIMINARY OUTLINE OF DISMANTLING STEPS

1. Disassemble non-essential process equipment, decontaminate and/or dispose as low level radioactive waste as appropriate.

2. Remove dry well tubes from shield tank; cut off portion of tank that protrudes above concrete shield; remove lead shielding in lower part of shield tank; place temporary decking over shield tank well to increase working space on top of concrete shield structure.

3. Remove the graphite thermal column as a single unit; construct shipping crate around graphite stack; prepare for shipment either to another licensee or to waste disposal site. (Note: the thermal column graphite is expected to be only slightly radioactive.)

4. Unstack lead bricks and graphite stringers from around the fuel boxes. Each new layer will be vacuumed with a HEPA filter equipped vacuum cleaner prior to unstacking. Personnel working on the graphite stack will wear half face dust respirators as further protection from any airborne radioactive dust. Air sampling equipment will be used during this operation to verify that airborne radioactivity levels are well below acceptable limits.

4a. Simultaneously with the unstacking operations, other personnel will package the graphite and lead in DOT approved shipping containers for subsequent disposal or shipment to another licensee. (If any of the filled containers generate significant radiation fields the removable concrete shield blocks will be used as temporary shielding.) Note: we plan to reserve several of the large sheets of lead from around the outside of the reactor graphite stack to potentially be used as temporary shielding both during step 5 and step 7.

5. Remove fuel boxes, flanged piping, control rods and shrouds and rod drive bearings and shafts. As much as possible, all such equipment will be removed intact. If cutting is necessary it will be done by sawing or shearing (not burning). All this material will be characterized as to type and quantity of radioactivity, packaged accordingly and disposed as low level radioactive waste. Note: with the possible exception of a few stainless steel parts we expect all of this material to be classifiable as LSA waste.

6. Assess the remaining radiation environment to determine whether it is feasible to conduct concrete core sampling of the biological shield and pedestal prior to removal of the rod drive support structure and lower portion of the shield tank.

7. Assuming concrete coring can safely be done at this point (perhaps using temporary lead shielding as discussed in step 4a), we will obtain cores for analysis to determine the thickness of concrete which must be removed from the interior of the biological shield to achieve DECON.

8. On the basis of the data obtained in step 7, we will prepare a "request for proposal" to solicit firm contractor bids for completion of DECON. At this point we will also be prepared to submit a final decommissioning plan to the NRC.

Note: during all of the operations described above a qualified health physicist will be present (full-time) to assure that appropriate contamination control procedures are used; to monitor all materials packaged for disposal or released for other use; to prepare appropriate records and shipping papers; and to generally assure that all radiation exposures are maintained ALARA.

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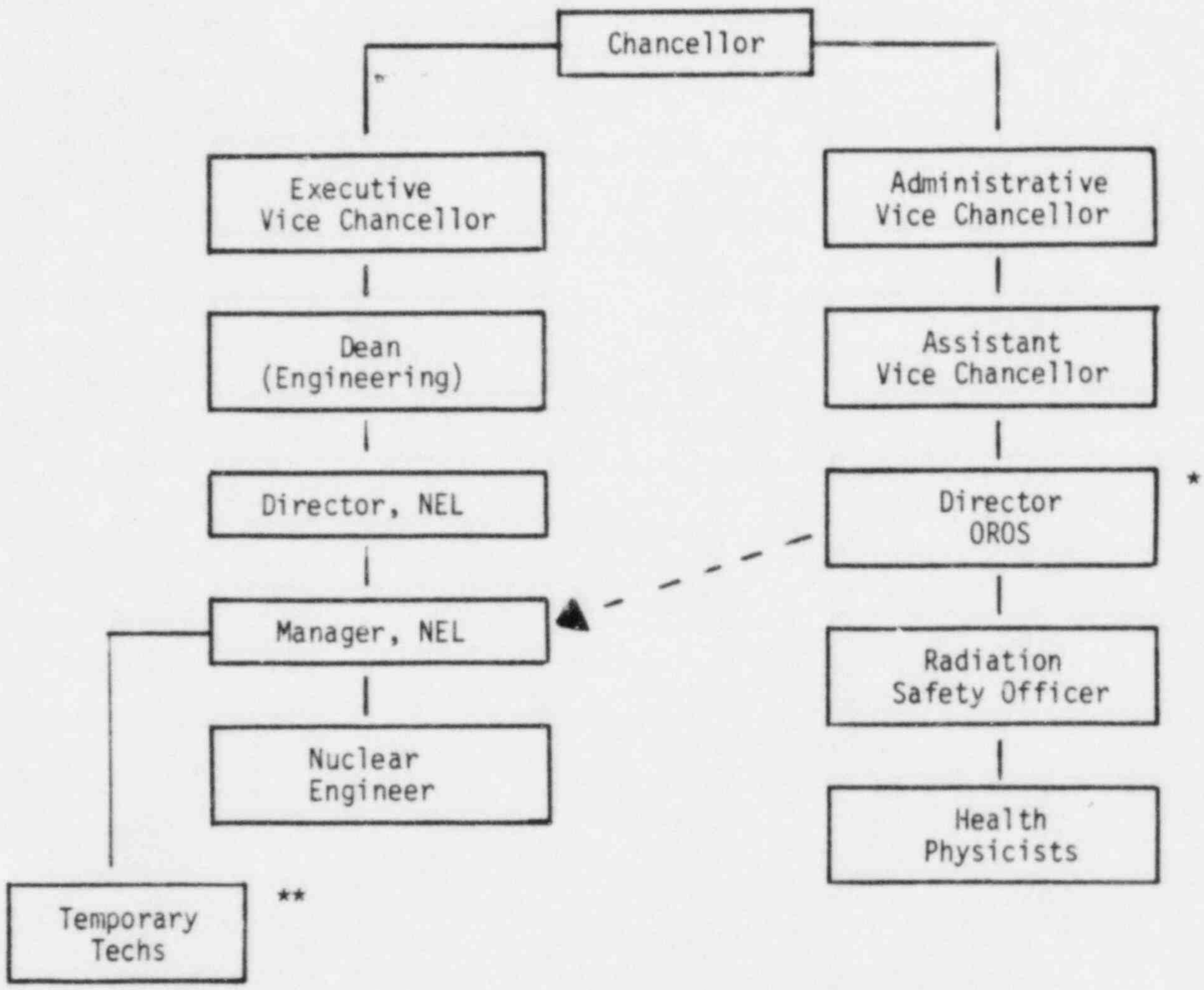
APPENDIX C

Dismantling Organization



* For the duration of the dismantling operations the manager of the NEL will take direction from the Director of OROS.

** The temporary technicians will be supervised by the manager of the NEL.